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# PUBLIC SUBMISSION

**Docket:** NRC-2010-0180
Availability of Draft NUREG-1800, Revision 2 and Draft NUREG-1801, Revision 2

**Comment On:** NRC-2010-0180-0001
Notice of Availability of Draft NUREG-1800, Revision 2; "Standard Review Plan for Review of License Renewal Applications for Nuclear Power Plants" and Draft NUREG-1801, Revision 2; "Generic Aging Lessons Learned (GALL) Report"

**Document:** NRC-2010-0180-DRAFT-0006
Comment on FR Doc # 2010-11841

## Submitter Information

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## General Comment

Please find attached industry comments for the Civil and Electrical portions of the GALL report. In addition, the Mechanical AMPs, with the exception of AMP M41, are also attached.

A subsequent submittal will be made to transmit to you the Mechanical AMP M41 and the Mechanical AMRs.

Julie Keys
NEI
Sr. Project Manager

## Attachments

**NRC-2010-0180-DRAFT-0006.1:** Comment on FR Doc # 2010-11841
**NRC-2010-0180-DRAFT-0006.2:** Comment on FR Doc # 2010-11841
**NRC-2010-0180-DRAFT-0006.3:** Comment on FR Doc # 2010-11841
**NRC-2010-0180-DRAFT-0006.4:** Comment on FR Doc # 2010-11841
**NRC-2010-0180-DRAFT-0006.5:** Comment on FR Doc # 2010-11841

SUNSI Review Complete
Template = ADM-013

E-RIDS = ADM-03
Add = R. Gramm (rag)

## X.M1 Metal Fatigue of the Reactor Coolant Pressure Boundary

Comment/Basis:

X.M1-1 Change the title to match the SCOPE of the program and Sections II through VIII of GALL. Cumulative fatigue damage exists for far more than the reactor coolant pressure boundary components as stated in the AMP. Containment, supports, steam generator secondary sides, reactor internals, ESF, Aux and S&P all have cumulative fatigue damage entries in the GALL AMR tables.

X.M1-2 Remove word "structural" and "reactor coolant system" from multiple locations so as not to unnecessarily restrict the program scope.

X.M1-3 The discussion of using NUREG/CR-6909 needs to be specific, not only in this program but in other portions of GALL Rev 2 and NUREG-1800 Rev 2. RG 1.207 and NUREG/CR-6909 do not allow use of the nickel-alloy Fen from 6909 with a CUF calculated from the existing ASME stainless steel curve. Assuming the staff wants to maintain this requirement, any discussion of this should be very specific. Suggest that rather than trying to summarize it here, just reference NUREGs 5704 and 6583 and 6909.

X.M1-4 The last paragraph of the program description is unnecessary. The GALL report wouldn't contain detailed description of a program that wasn't acceptable.

X.M1-5 As the scope now applies to more than the RCS, separate the RCS environmental fatigue to a second paragraph in the SCOPE. Include some of the words from DETECTION OF AGING EFFECTS as they fit better here.

X.M1-6 In Detection of Aging Effects need to delete the discussion of monitoring specific locations. Add some of it in the scope as discussed above. Note that this should be an option, as it is in PARAMETERS MONITORED/INSPECTED, not a requirement. Most plants just count cycles rather than monitor specific locations.

X.M1-7 Need to reword MONITORING AND TRENDING, ACCEPTANCE CRITERIA, and CORRECTIVE ACTIONS to be more concise and not to mention the RCS pressure boundary.

X.M1-8 In Program Description recommend add the following in the First paragraph - at end: "The program also verifies that the severity of the monitored transients are bounded by the design transient definition for which they are classified." To provide additional clarity on transients.

X.M1-9 Recommend adding the following in Program Description in the last paragraph to provide clarification on the use of Fen. "The environmentally-adjusted Cumulative Usage Factor is calculated by multiplying the Cumulative Usage Factor (CUF) by an environmental correction factor, Fen. The environmental correction factor for carbon or low-alloy steel may be computed using the equations from either NUREG/CR-6583 or NUREG/CR-6909, applied to CUF value determined using the applicable

ASME Section III fatigue curve. The environmental correction factor for austenitic stainless steel may be computed using the equation from NUREG/CR-5704 in conjunction with the CUF value determined using the ASME Section III fatigue curve. Alternatively, the environmental correction factor for austenitic stainless steel may be determined using the equation from NUREG/CR-6909 in conjunction with the CUF value determined using either the NUREG/CR-6909 fatigue curve or the ASME Section III fatigue curve. The environmental correction factor for nickel-based alloys may be computed using the equation from NUREG/CR-6909 in conjunction with the CUF value determined using the NUREG/CR-6909 stainless steel fatigue curve."

X.M1-10 Scope of program, add: The scope includes those components that have been identified to have a fatigue TLAA.

X.M1-11 Preventive Actions: For clarity change to: "The program prevents the fatigue TLAAs from becoming invalid by assuring that the fatigue usage resulting from actual operational transients does not exceed the Code design limit of 1.0, including environmental effects where applicable. This could be caused by the numbers of actual plant transients exceeding the numbers used in the fatigue analyses or by the actual transient severity exceeding the bounds of the design transient definitions. However, in either of these cases, if the analysis is revised to account for the increased number or severity of transients such that the CUF value remains below 1.0, the program remains effective."

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

### Program Description

Fatigue usage factor is a computed mechanical parameter suitable for gauging fatigue damage in ~~structural~~ components subjected to fluctuating stresses. Crack initiation is assumed to have started in a ~~structural~~ component when the fatigue usage factor at a point of the ~~structural~~ component reaches the value of 1, the design limit on fatigue. 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 program also verifies that the severity of the monitored transients are bounded by the design transient definition for which they are classified.

The AMP addresses the effects of the reactor coolant environment on component fatigue life by assessing the impact of the reactor coolant environment on a set of sample critical components for the plant. Examples of critical components are identified in NUREG/CR-6260. ~~The environmentally adjusted fatigue usage factor can be calculated by multiplying the regular fatigue usage factor evaluated in accordance with the ASME Section III guidelines, in air environment, by an environmental fatigue life correction factor~~. The environmentally-adjusted Cumulative Usage Factor is calculated by multiplying the Cumulative Usage Factor (CUF) by an environmental correction factor, Fen. The environmental correction factor for carbon or low-alloy steel may be computed using the equations from either NUREG/CR-6583 or NUREG/CR-6909, applied to CUF value determined using the applicable ASME Section III fatigue curve. The environmental correction factor for austenitic stainless steel may be computed using the equation from NUREG/CR-5704 in conjunction with the CUF value determined using the ASME Section III fatigue curve. Alternatively, the environmental correction factor for austenitic stainless steel may be determined using the equation from NUREG/CR-6909 in conjunction with the CUF value determined using either the NUREG/CR-6909 fatigue curve or the ASME Section III fatigue curve. The environmental correction factor for nickel-based alloys may be computed using the equation from NUREG/CR-6909 in conjunction with the CUF value determined using the NUREG/CR-6909 stainless steel fatigue curve. Formulae for calculating the environmental fatigue life correction factors are contained in NUREG/CR-5704 for stainless steel, in NUREG/CR-6583 for carbon and low alloy steels, and in NUREG/CR-6909 for carbon and low alloy steel, stainless steel, and nickel alloys.

~~As discussed below, this is an acceptable program for managing metal fatigue for the reactor coolant pressure boundary, considering environmental effects.~~

**Evaluation and Technical Basis**

1. ***Scope of Program:*** The scope includes those components that ~~the GALL Report identifies in AMR line items that have an aging effect of cumulative fatigue damage~~ have been identified to have a fatigue TLAA. The program monitors and tracks the number of critical thermal and pressure transients for the selected ~~reactor coolant system~~ components. ~~For a set of sample of components, the program includes fatigue usage calculations that consider the effects of the reactor water environment.~~ The program ensures the fatigue usage remaining within the allowable limit, thus minimizing fatigue cracking of metal components ~~of the reactor coolant pressure boundary~~ caused by anticipated cyclic strains in the material.

   For a set of sample reactor coolant system components, the program includes fatigue usage calculations that consider the effects of the reactor water environment. This sample set includes the locations identified in NUREG/CR-6260, or proposes alternative locations based on plant configuration.

2. ***Preventive Actions:*** ~~The program consists of transient monitoring and tracking mechanisms to measure the severity of transient events and number of occurrences so as to prevent the cumulative usage factor from exceeding the design code limits.~~ The program prevents the fatigue TLAAs from becoming invalid by assuring that the fatigue usage resulting from actual operational transients does not exceed the Code design limit of 1.0, including environmental effects where applicable. This could be caused by the numbers of actual plant transients exceeding the numbers used in the fatigue analyses or by the actual transient severity exceeding the bounds of the design transient definitions. However, in either of these cases, if the analysis is revised to account for the increased number or severity of transients such that the CUF value remains below 1.0, the program remains effective.

3. ***Parameters Monitored/Inspected:*** The program monitors all plant design transients that cause cyclic strains, which are significant contributors to the fatigue usage factor. The number of occurrences of the plant transients that cause significant fatigue usage for each ~~critical reactor coolant pressure boundary~~ component is to be monitored. Alternatively, more detailed monitoring of local pressure and thermal conditions may be performed to allow the actual fatigue usage for the specified critical locations to be calculated.

4. ***Detection of Aging Effects:*** The program provides for updates of the fatigue usage calculations on an as-needed basis if an allowable cycle limit is approached, or in a case where a transient definition has been changed, unanticipated new thermal events are discovered, or the geometry of components have been modified. ~~The program monitors a~~

~~set of sample high fatigue usage locations. This sample set includes the locations identified in NUREG/CR-6260, as minimum, or proposes alternatives based on plant configuration.~~

5. ***Monitoring and Trending:*** Trending is assessed to ensure that the fatigue usage factor ~~tends to be confined within the~~remains below the design ~~allowable~~ limit during the period of extended operation, thus minimizing fatigue cracking of metal components ~~of the reactor coolant pressure boundary~~ caused by anticipated cyclic strains in the material.

6. ***Acceptance Criteria:*** The acceptance criterion is maintaining the cumulative fatigue usage below the design ~~code~~ limit through the period of extended operation ~~renewed license term~~, with consideration of the reactor water environmental fatigue effects described in the program description and *Scope of the Program*.

7. ***Corrective Actions:*** The program provides for corrective actions to prevent the usage factor from exceeding the design ~~code~~ limit during the period of extended operation. Acceptable corrective actions include repair of the component, replacement of the component, and a more rigorous analysis of the component to demonstrate that the design ~~code~~ limit will not be exceeded during the period of extended operation. For programs that monitor high fatigue usage locations, corrective actions include a review of additional affected ~~reactor coolant pressure boundary~~ locations. 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:*** Site quality assurance procedures, review and approval processes, and administrative controls are implemented in accordance with the requirements of Appendix B to 10 CFR Part 50. 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 and administrative controls.

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

10. ***Operating Experience:*** The program reviews industry experience relevant to fatigue cracking. Applicable operating experience relevant to fatigue cracking is to be considered in selecting the locations for monitoring. As discussed in NRC Regulatory Issue Summary 2008-30, the use of certain simplified analysis methodology to demonstrate compliance with the ASME Code fatigue acceptance criteria could be non-conservative; therefore, a confirmatory analysis is recommended.

**References**

NRC Regulatory Issue Summary 2008-30, *Fatigue Analysis of Nuclear Power Plant Components*, U.S. Nuclear Regulatory Commission, December 16, 2008.

NUREG/CR-6260, *Application of NUREG/CR-5999 Interim Fatigue Curves to Selected Nuclear Power Plant Components,* U.S. Nuclear Regulatory Commission, March 1995.

NUREG/CR-6909, *Effects of LWR Coolant Environments on the Fatigue Life of Reactor Materials,* U.S. Nuclear Regulatory Commission, February 2007.

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**XI.M2 Water Chemistry**

Comment/Basis:

XI M2-1 Program Description – Do not delete the "or later revisions" wording that was added during Revision 1 of the GALL. The references to BWRVIP-190, EPRI 1014986, Rev. 6 & 1016555, Rev. 7 are good changes and it acknowledges the most current industry guidance to right now, but these documents are revised every few years based on industry experience and plants implement the most current guidance. Therefore if you remove the "or later revisions" wording, the GALL will quickly become out of date. Suggest adding "(reviewed and accepted by the NRC in a safety evaluation report)" to allow versions reviewed during inspections that confirm the adequacy of the later versions and could eliminate exception.

XI M2-2 Program Description, Elements1, 2 & 3 - Delete the specific parameters that are to be monitored and just reference the EPRI water chemistry guidelines. In several cases this causes contradictions to the guidelines and results in exceptions to the program.

Examples:

1) Element 3 indicates that hydrogen peroxide is monitored to mitigate degradation of structural materials. However this contradicts the guidance in BWRVIP-190. Rapid decomposition of hydrogen peroxide makes reliable data difficult to obtain and BWRVIP-190 Section 6.3.3, "Water Chemistry Guidelines for Power Operation," does not address monitoring for hydrogen peroxide. Noble metal chemical application and hydrogen addition are generally used to mitigate occurrence of IGSCC of structural materials by suppressing the formation of hydrogen peroxide. The hydrogen addition generally accomplishes an Electrochemical Corrosion Potential (ECP) value less than -230mV, SHE (Standard Hydrogen Electrode). By maintaining a low ECP less than -230mV, SHE, the reactor water chemistry minimizes the effects from hydrogen peroxide below the threshold that prompted the issue raised in NUREG 1801. In addition the ISI program investigates structural degradation in potentially affected locations and provides condition monitoring of the reactor vessel, reactor internal components and ASME Class 1 pressure retaining components in accordance with ASME Section XI, Subsection IWB. Indications and relevant conditions detected during examinations are evaluated in accordance with ASME Section XI Articles IWB-3000, for Class 1.

2) Element 3 indicates that dissolved oxygen is monitored; however BWRVIP-190 acknowledges the difficulty with monitoring dissolved oxygen and sets limits for conductivity, chlorides, sulfates and total organic carbon (TOC) as an alternate method for ensuring component integrity.

3) Program Description, Elements1, 2, 3 indicates that water quality (pH and conductivity) is maintained in accordance with established guidance. However, BWRVIP-190, Section 8.3.4.5, indicates pH measurement accuracy in most BWR streams is unreliable because of the dependence of the instrument reading on ionic strength of the sample solution. In addition, the monitoring of pH is not discussed in BWRVIP-190, Appendix E for condensate storage tank, demineralized water storage tank, or torus water.

XI M2-3 Element 7 – Remove the "root" in root cause identified. In many cases the root cause of the unacceptable chemistry results may not be able to be identified. The more important investigation should be to determine if the excursion affected the components. In addition the individual corrective action processes should determine if root cause identification is required.

## XI.M2 WATER CHEMISTRY

### Program Description

The main objective of this program is to mitigate loss of material due to corrosion, cracking due to stress corrosion cracking (SCC) and related mechanisms, and reduction of heat transfer due to fouling in components exposed to a treated water environment. The program includes periodic monitoring of the treated water and control of known detrimental contaminants in order to minimize ~~such as chlorides, fluorides (pressurized water reactors [PWRs] only), dissolved oxygen, and sulfate concentrations below the levels known to result in~~ loss of material or cracking.

The water chemistry program for boiling water reactors (BWRs) relies on monitoring and control of reactor water chemistry based on industry guidelines, such as the Boiling Water Reactor Vessel and Internals Project (BWRVIP)-190 (Electric Power Research Institute [EPRI] 1016579) or later revisions (reviewed and accepted by the NRC in a safety evaluation report). The BWRVIP-190 has three sets of guidelines: one for primary water, one for condensate and feedwater, and one for control rod drive (CRD) mechanism cooling water. The water chemistry program for PWRs relies on monitoring and control of reactor water chemistry based on industry guidelines such as EPRI 1014986 (PWR Primary Water Chemistry Guidelines-Revision 6) and EPRI 1016555 (PWR Secondary Water Chemistry Guidelines-Revision 7) or later revisions (reviewed and accepted by the NRC in a safety evaluation report).

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

### Evaluation and Technical Basis

**1.** ***Scope of Program:*** The program includes specifications for chemical species, impurities and additives, sampling and analysis frequencies, and corrective actions for control of reactor water chemistry. System water chemistry is controlled to minimize contaminant concentration and mitigate loss of material due to general, crevice, and pitting corrosion and cracking caused by SCC. For BWRs, maintaining high water purity reduces susceptibility to SCC, and chemical additive programs such as hydrogen water chemistry, or noble metal chemical application also may be used. For PWRs, additives are used for reactivity control and to control pH and inhibit corrosion.

**2.** ***Preventive Actions:*** The program includes specifications for chemical species, impurities and additives, sampling and analysis frequencies, and corrective actions for control of reactor water chemistry. System water chemistry is controlled to minimize contaminant concentration and mitigate loss of material due to general, crevice, and pitting corrosion and cracking caused by SCC. For BWRs, maintaining high water purity reduces susceptibility to SCC, and chemical additive programs such as hydrogen water chemistry, or noble metal chemical application also may be used. For PWRs, additives are used for reactivity control and to control pH and inhibit corrosion.

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

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

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

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

4. ***Detection of Aging Effects:*** This is a mitigation program and does not provide for detection of any aging effects for the components within its scope. The monitoring methods and frequency of water chemistry sampling and testing is performed in accordance with the EPRI water chemistry guidelines and based on plant operating conditions. The main objective of this program is to mitigate loss of material due to corrosion and by cracking due to SCC in components exposed to a treated water environment.

5. ***Monitoring and Trending:*** Chemistry parameter data are recorded, evaluated, and trended in accordance with the EPRI water chemistry guidelines.

6. ***Acceptance Criteria:*** Maximum levels for various chemical parameters are maintained within the system-specific limits as indicated by the limits specified in the corresponding EPRI water chemistry guidelines.

7. ***Corrective Actions:*** Any evidence of aging effects or unacceptable water chemistry results are evaluated, the ~~root~~ cause identified, and the condition corrected. When measured water chemistry parameters are outside the specified range, corrective actions are taken to bring the parameter back within the acceptable range (or to change the operational mode of the plant) within the time period specified in the EPRI water chemistry guidelines. Whenever corrective actions are taken to address an abnormal chemistry condition, increased sampling or other appropriate actions may be used to verify the effectiveness of these actions. 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:*** Following corrective actions, additional samples are taken and analyzed to verify that the corrective actions were effective in returning the concentrations of contaminants, such as chlorides, fluorides, sulfates, dissolved oxygen, and hydrogen peroxide, to within the acceptable ranges. As discussed in the appendix to this report, the staff finds the requirements of 10 CFR Part 50, Appendix B, acceptable to address the confirmation process.

9. ***Administrative Controls:*** Site quality assurance 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 administrative controls.

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

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

*PWR Primary System:* The potential for SCC-type mechanisms might normally occur because of inadvertent introduction of contaminants into the primary coolant system, including contaminants introduced from the free surface of the spent fuel pool (which can be a natural collector of airborne contaminants) or the introduction of oxygen during plant cooldowns (NRC IN 84–18). Ingress of demineralizer resins into the primary system has caused IGSCC of Alloy 600 vessel head penetrations (NRC IN 96-11, NRC GL 97-01). Inadvertent introduction of sodium thiosulfate into the primary system has caused IGSCC of steam generator tubes. SCC has occurred in safety injection lines (NRC INs 97-19 and 84-18), charging pump casing cladding (NRC INs 80-38 and 94-63), instrument nozzles in safety injection tanks (NRC IN 91-05), and safety-related SS piping systems that contain oxygenated, stagnant, or essentially stagnant borated coolant (NRC IN 97-19). Steam generator tubes and plugs and Alloy 600 penetrations have experienced primary water stress corrosion cracking (NRC INs 89-33, 94-87, 97-88, 90-10, and 96-11; NRC Bulletin 89-01 and its two supplements). IGSCC-induced circumferential cracking has occurred in PWR pressurizer heater sleeves (NRC IN 2006-27).

*PWR Secondary System:* Steam generator tubes have experienced ODSCC, IGA, wastage, and pitting (NRC IN 97-88, NRC GL 95-05). Carbon steel support plates in steam generators have experienced general corrosion. The steam generator shell has experienced pitting and stress corrosion cracking (NRC INs 82-37, 85-65, and 90-04). Extensive buildup of deposits at steam generator tube support holes can result in flow-induced vibrations and tube cracking (NRC IN 2007-37).

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

NRC Information Notice 2006-27, *Circumferential Cracking in the Stainless Steel Pressurizer Heater Sleeves of Pressurized Water Reactors*, December 11, 2006.

NRC Information Notice 2007-37, *Buildup of Deposits in Steam Generators*, November 23, 2007.

NUREG-1544, *Status Report: Intergranular Stress Corrosion Cracking of BWR Core Shrouds and Other Internal Components*, U.S. Nuclear Regulatory Commission, March 1, 1996.

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## XI.M3 REACTOR HEAD CLOSURE STUD Bolting

Comment/Basis:

XI.M3-1 Suggest rewording PREVENTIVE ACTIONS to make each action a bullet, rather than 2 bullets and 2 actions in the text. See below

## XI.M3 REACTOR HEAD CLOSURE STUD Bolting

### Program Description

This program includes (a) inservice inspection (ISI) in accordance with the requirements of the American Society of Mechanical Engineers (ASME) Code, *Section XI, Subsection* IWB (2004 edition2[1], no addenda), Table IWB 2500-1; and (b) preventive measures to mitigate cracking. The program also relies on recommendations to address reactor head stud bolting degradation as delineated in NUREG-1339 and Nuclear Regulatory Commission (NRC) Regulatory Guide (RG) 1.65.

### Evaluation and Technical Basis

1. ***Scope of Program:*** The program manages the aging effects of cracking due to stress corrosion cracking (SCC) or intergranular stress corrosion cracking (IGSCC) and loss of material due to wear or corrosion for reactor vessel closure stud bolting (studs, washers, bushings, nuts, and threads in flange) for both boiling water reactors (BWRs) and pressurized water reactors (PWRs).

2. ***Preventive Actions:*** Preventive measures include

   (a) avoiding the use of metal-plated stud bolting to prevent degradation due to corrosion or hydrogen embrittlement, ~~and~~
   (b) using manganese phosphate or other acceptable surface treatments, ~~and~~
   (c) using stable lubricants. ~~(RG 1.65).~~ Of particular note, use of molybdenum disulfide ($MoS_2$) as a lubricant has been shown to be a potential contributor to SCC and should not be used. (RG 1.65).
   (d) ~~Preventive measures also include~~ using bolting material for closure studs that has an actual measured yield strength ~~limited to~~ less than 1,034 megapascals (MPa) (150 kilo-pounds per square inch) (NUREG-1339).

   Implementation of these mitigation measures can reduce potential for SCC or IGSCC, thus making this program effective.

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[1] Refer to the GALL Report, Chapter I, for applicability of other editions of the ASME Code, Section XI.

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

4. ***Detection of Aging Effects:*** The extent and schedule of the inspection and test techniques prescribed by the program are designed to maintain structural integrity and ensure that aging effects are 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.

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

   Components are examined and tested in accordance with ASME Code, Section XI, Table IWB-2500-1, examination category B-G-1, for pressure-retaining bolting greater than 2 inches in diameter. Examination category B-P for all pressure-retaining components specifies visual VT-2 examination of all pressure-retaining boundary components during the NUREG-1801, Rev. 2 XI M3-2 April 2010 DRAFT system leakage test and the system hydrostatic test. Table IWB-2500-1 specifies the extent and schedule of the inspection and examination methods.

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

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

7. ***Corrective Actions:*** Repair and replacement are performed in accordance with the requirements of IWA-4000 and the material and inspection guidance of RG 1.65. Maximum yield strength of replacement material should be limited as recommended in NUREG-1339. 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:*** Site quality assurance 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 confirmation process and administrative controls.

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:*** SCC has occurred in BWR pressure vessel head studs (Stoller, 1991). The aging management program has provisions regarding inspection techniques and evaluation, material specifications, corrosion prevention, and other aspects of reactor pressure vessel head stud cracking. Implementation of the program provides reasonable assurance that the effects of cracking due to SCC or IGSCC and loss of material due to

wear are adequately managed so that the intended functions of the reactor head closure studs and bolts are maintained consistent with the current licensing basis for the period of extended operation. Degradation of threaded bolting and fasteners in closures for the reactor coolant pressure boundary has occurred from boric acid corrosion, SCC, and fatigue loading (NRC Inspection and Enforcement Bulletin 82-02, NRC Generic Letter 91-17).

**References**

10 CFR Part 50, Appendix B, *Quality Assurance Criteria for Nuclear Power Plants*, Office of the Federal Register, National Archives and Records Administration, 2009.

10 CFR 50.55a, *Codes and Standards*, Office of the Federal Register, National Archives and Records Administration, 2009.

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## XI.M4 BWR VESSEL ID ATTACHMENT WELDS

Comment/Basis:

XI.M4-1 The PREVENTIVE ACTIONS element varies widely among several condition monitoring programs (XI.M4, XI.M7, XI.M8, XI.M9, XI.M11B, etc.) that have no preventive actions, including the discussion of preventive actions in the Water Chemistry program. This wording should be consistent for all the involved programs. See below for suggested best wording. Correct corespray to core spray Program Description

## XI.M4 BWR VESSEL ID ATTACHMENT WELDS

### Program Description

The program includes inspection and flaw evaluation in accordance with the guidelines of staff-approved boiling water reactor vessel and internals project (BWRVIP-48A) to ensure the long-term integrity and safe operation of boiling water reactor (BWR) vessel inside diameter (ID) attachment welds.

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

### Evaluation and Technical Basis

1. ***Scope of Program:*** The program is focused on managing the effects of cracking due to stress *corrosion cracking (SCC), including intergranular stress corrosion cracking* (IGSCC). The program is an augmented inservice inspection program that uses the inspection and flaw evaluation criteria in BWRVIP-48A to detect cracking and monitor the effects of cracking on the intended function of the components. The program provides for repair and/or replacement, as needed, to maintain the ability to perform the intended function. The

program is applicable to structural welds for BWR reactor vessel internal integral attachments.

2. ***Preventive Actions:*** The BWR Vessel ID Attachment Welds Program is a condition monitoring Program and has no preventive actions.

   ~~VIP-48A provides guidance on detection but does not provide guidance on methods to mitigate cracking.~~ Maintaining high water purity reduces susceptibility to SCC or IGSCC. Reactor coolant water chemistry is monitored and maintained in accordance with the Water Chemistry Program. ~~the guidelines in BWRVIP-130 (EPRI TR-1008192) or later revisionsThe water chemistry program for BWRs relies on monitoring and control of reactor water chemistry based on industry guidelines, such as BWRVIP-190 (Electric Power Research Institute [EPRI] 1016579). BWRVIP-190 has three sets of guidelines: one for primary water, one for condensate and feedwater, and one for control rod drive (CRD) mechanism cooling water~~. The program description, evaluation, and technical basis of ~~monitoring and maintaining reactor~~ water chemistry are presented in Section XI.M2, "Water Chemistry."

3. ***Parameters Monitored/Inspected:*** The program monitors for cracks induced by SCC and IGSCC on the intended function of BWR vessel ID attachment welds. The program looks for surface discontinuities that may indicate the presence of a crack in the component in accordance with the guidelines of approved BWRVIP-48A and the requirements of the American Society of Mechanical Engineers (ASME) Code, Section XI, Table IWB 2500-1 (2004 edition[2]).

4. ***Detection of Aging Effects:*** The extent and schedule of the inspection and test techniques prescribed by BWRVIP-48A guidelines are designed to maintain structural integrity and ensure that aging effects are discovered and repaired before the loss of intended function. Inspection can reveal cracking. Vessel ID attachment welds are inspected in accordance with the requirements of ASME Section XI, Subsection IWB, Examination Category B-N-2. The ASME Code, Section XI inspection specifies visual VT-1 examination to detect discontinuities and imperfections on the surfaces of components and visual VT-3 examination to determine the general mechanical and structural condition of the component supports. The inspection and evaluation guidelines of BWRVIP-48A recommend more stringent inspections for certain attachments. The guidelines recommend enhanced visual VT-1 examination of all safety-related attachments and those non-safety-related attachments identified as being susceptible to IGSCC. Visual VT-1 examination is capable of achieving 1/32-inch resolution; the enhanced visual VT-1 examination method is capable of achieving a 1-millimeter wire resolution. The nondestructive examination (NDE) techniques appropriate for inspection of BWR vessel internals, including the uncertainties inherent in delivering and executing NDE techniques in a BWR, are included in BWRVIP-03.

5. ***Monitoring and Trending:*** Inspections scheduled in accordance with ASME Code, Section XI, IWB-2400 and approved BWRVIP-48A guidelines provide timely detection of cracks. If flaws are detected, the scope of examination is expanded. Any indication detected is evaluated in accordance with ASME Code, Section XI or the staff-approved BWRVIP-48A guidelines. Applicable and approved BWRVIP-14A, BWRVIP-59A, and BWRVIP-60A

[2] Refer to the GALL Report, Chapter I, for applicability of other editions of the ASME Code, Section XI.

documents provide guidelines for evaluation of crack growth in stainless steels, nickel alloys, and low-alloy steels, respectively.

6. ***Acceptance Criteria:*** Acceptance criteria are given in BWRVIP-48A and ASME Code, Section XI.

7. ***Corrective Actions:*** Repair and replacement procedures are equivalent to those requirements in ASME Code, Section XI. Corrective action is performed in accordance with ASME Code, Section XI, IWA-4000. As discussed in the appendix to this report, the staff finds that licensee implementation of the corrective action guidelines in BWRVIP-48A provides an acceptable level of quality in accordance with 10 CFR Part 50, Appendix B corrective actions.

8. ***Confirmation Process:*** Site quality assurance procedures, review and approval processes, and administrative controls are implemented in accordance with the requirements of 10 CFR Part 50, Appendix B. As discussed in the appendix to this report, the staff finds that licensee implementation of the guidelines in BWRVIP-48A provides an acceptable level of quality in accordance with the 10 CFR Part 50, Appendix B confirmation process and administrative controls.

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

**References**

10 CFR Part 50, Appendix B, *Quality Assurance Criteria for Nuclear Power Plants*, Office of the Federal Register, National Archives and Records Administration, 2009.

10 CFR 50.55a, *Codes and Standards*, Office of the Federal Register, National Archives and Records Administration, 2009.

ASME Section XI, *Rules for Inservice Inspection of Nuclear Power Plant Components*, The ASME Boiler and Pressure Vessel Code, 2004 edition as approved in 10 CFR 50.55a, The American Society of Mechanical Engineers, New York, NY.

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

BWRVIP-14A (EPRI 1016569), *Evaluation of Crack Growth in BWR Stainless Steel RPV Internals*, September 2008.

BWRVIP-48A (EPRI 1009948), *BWR Vessel and Internals Project, Vessel ID Attachment Weld Inspection and Flaw Evaluation Guidelines*, November 2004.

BWRVIP-59A (EPRI 1014874), *Evaluation of Crack Growth in BWR Nickel-Base Austenitic Alloys in RPV Internals*, May 2007.

BWRVIP-60A (EPRI 1008871), *BWR Vessel and Internals Project, Evaluation of Crack Growth in BWR Low Alloy Steel RPV Internals*, June 2003.

BWRVIP-62 (EPRI 108705), *BWR Vessel and Internals Project, Technical Basis for Inspection Relief for BWR Internal Components with Hydrogen Injection*, March 7, 2000.

BWRVIP-190 (EPRI 1016579), *BWR Vessel and Internals Project: BWR Water Chemistry Guidelines—2008 Revision*, October 2008.

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## XI.M6 BWR CONTROL ROD DRIVE RETURN LINE NOZZLE

Comment/Basis

XI.M6 - 1 Remove extra wording in Scope. There is no (b) option in the new wording.

XI.M6 – Typo – Element 4 change blend to bend and program description change corosion to corrosion

XI.M6 – 3 Add "reviewed and accepted by the NRC in a safety evaluation report" after later revisions to match earlier changes to allow the use of later revisions.

**XI.M6 BWR CONTROL ROD DRIVE RETURN LINE NOZZLE**
**Program Description**
This program is a condition monitoring program for boiling water reactor (BWR) control rod drive return line (CRDRL) nozzles that is based on the staff's recommended position in NUREG-0619 for thermal fatigue. This program is also intended to address stress corrosion cracking (SCC) discussed in IN 2004-08. The augmented inspections performed in accordance with the recommendations in NUREG-0619 supplement those in-service inspections that are required for these nozzles in accordance with the American Society of Mechanical Engineers (ASME) Code, Section XI, Table IWB-2500-1, as mandated through reference in 10 CFR 50.55a. Thus, this program includes (a) mandatory in-service inspection (ISI) in accordance with the ASME Code, Section XI, Table IWB 2500-1 (2004 edition1[5]), and (b) augmented ISI examinations in accordance with applicant's commitments to U.S. Nuclear Regulatory Commission (NRC) Generic Letter (GL) 80-095 to implement the recommendations in NUREG-0619.

5 Refer to the GALL Report, Chapter I, for applicability of other editions of the ASME Code, Section XI.

**Evaluation and Technical Basis**

1. ***Scope of Program:*** The program manages the effects of cracking on the intended pressure boundary function of CRDRL nozzles. The scope of this program is applicable to BWRs whose reactor vessel (RV) design includes a welded CRDRL nozzle design. The scope of the program includes CRDRL nozzles and their nozzle-to-RV welds, which are ASME Code Class 1 components. The scope of the program also includes a CRDRL nozzle cap (including any CRDRL nozzle-to-cap welds) if, to mitigate cracking, an applicant has ~~either: (a)~~ cut the piping to the CRDRL nozzle, and capped the CRDRL nozzle.

2. ***Preventive Actions:*** Activities for preventing or mitigating cracking in CRDRL nozzles are consistent with a BWR facility's past preventive or mitigation actions/activities in its current licensing basis as stated in the applicant's docketed response to NRC GL 80-095 and made to address the recommendations in NUREG-0619. Maintaining high water purity reduces susceptibility to SCC. The water chemistry program for BWRs relies on monitoring and control of reactor water chemistry based on industry guidelines, such as BWRVIP-190 (Electric Power Research Institute [EPRI] 1016579) or later revisions(reviewed and accepted by the NRC in a

safety evaluation report). BWRVIP-190 has three sets of guidelines: one for primary water, one for condensate and feedwater, and one for control rod drive (CRD) mechanism cooling water. The program description and evaluation and technical basis of monitoring and maintaining reactor water chemistry are addressed through implementation of Section XI.M2, "Water Chemistry."

3. ***Parameters Monitored/Inspected:*** The aging management program (AMP) manages the effects of cracking on the intended function of the RV, the CRDRL nozzle, and for capped nozzles, the nozzle caps and cap-to-nozzle welds. For liquid penetrant test (PT) examinations that are implemented in accordance with this AMP, the AMP monitors for linear indications that may be indicative of surface breaking cracks. For the volumetric ultrasonic test (UT) examinations that are performed in accordance with this AMP, the AMP monitors and evaluates signals that may indicate the presence of a planar flaw (crack).

4. ***Detection of Aging Effects:*** The extent and schedule of inspection, as delineated in NUREG-0619, assures detection of cracks before the loss of intended function of the CRDRL nozzles. Inspection and test recommendations include PT of CRDRL nozzle blend radius and bore regions and the RV wall area beneath the nozzle, control rod drive system performance testing, and for capped nozzles, the nozzle caps and cap-to-nozzle welds. The inspection is to include base metal to a distance of one-pipe-wall thickness or 0.5 inches, whichever is greater, on both sides of the weld.

5. ***Monitoring and Trending:*** The inspection schedule of NUREG-0619 provides timely detection of cracks. Indications of cracking are evaluated and trended in accordance with the ASME Code, Section XI, IWB-3100, against applicable acceptance standard criteria that are specified in the ASME Code, Section XI, IWB-3400 or IWB-3500.

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

7. ***Corrective Actions*** Corrective action is performed in conformance with ASME Code, Section XI, IWA-4000. 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:*** Site quality assurance 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 confirmation process and administrative controls.

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:*** Cracking of CRDRL nozzle-to-vessel and nozzle-to-cap welds has occurred in several BWR plants (NUREG-0619 and Information Notice 2004-08). The present AMP has been implemented for nearly 25 years and has been found to be effective in managing the effects of cracking on the intended function of CRDRL nozzles.

**References**

10 CFR Part 50, Appendix B, *Quality Assurance Criteria for Nuclear Power Plants*, Office of the Federal Register, National Archives and Records Administration, 2009.

10 CFR 50.55a, *Codes and Standards,* Office of the Federal Register, National Archives and Records Administration, 2009.
ASME Section XI, *Rules for Inservice Inspection of Nuclear Power Plant Components*, The ASME Boiler and Pressure Vessel Code, 2004 edition as approved in 10 CFR 50.55a, The American Society of Mechanical Engineers, New York, NY.
Letter from D. G. Eisenhut, U.S. Nuclear Regulatory Commission, to R. Gridley, General Electric Company, *forwarding NRC Generic Technical Activity A-10*, January 28, 1980.April 2010 DRAFT XI M6-3 NUREG-1801, Rev. 2

NRC Generic Letter 80-095, (Untitled), November 13, 1980.6
NRC Generic Letter 81-11, (Untitled), February 29, 1981.7
NRC Information Notice 2004-08, *Reactor Coolant Pressure Boundary Leakage Attributable To Propagation of Cracking In Reactor Vessel Nozzle Welds*, U.S. Nuclear Regulatory Commission, April 22, 2004.

NUREG-0619, *BWR Feedwater Nozzle and Control Rod Drive Return Line Nozzle Cracking*,

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## XI.M7 BWR STRESS CORROSION CRACKING

Comment/Basis:

XI.M7-1 It seems odd that the ASME Code isn't referred to when discussing the inspections to be performed in elements 1, 3, or 4. Then suddenly, the results are trended per the Code in element 5, have acceptance criteria per the code in element 6, and corrective action per the code in element 7. Shouldn't the inspections required by the code be mentioned earlier on? (Mark-up not provided)

XI.M7-2 Again, the PREVENTIVE ACTIONS section should be consistent with other condition monitoring AMPs. (See comment on AMP XI.M4)

### 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 comprehensive program outlined in NUREG-0313, Rev 2 and NRC GL 88 01 describe improvements that, in combination, will reduce the susceptibility to IGSCC. These elements consist of a susceptible (sensitized) material, a significant tensile stress, and an aggressive environment. Sensitization of nonstabilized austenitic stainless steels containing greater than 0.03 weight percent carbon involves precipitation of chromium carbides at the grain boundaries during certain fabrication or welding processes. The formation of carbides creates a chromium depleted region that, in certain environments, is susceptible to stress corrosion cracking (SCC). Residual tensile stresses are introduced from fabrication processes, such as welding, surface grinding, or forming. High levels of dissolved oxygen or aggressive contaminants, such as sulfates or chlorides, accelerate the SCC processes. The program 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-75A) report allows for modifications to the inspection scope in the GL 88-01 program.

## Evaluation and Technical Basis

1. ***Scope of Program:*** The program focuses on (a) managing and implementing countermeasures to mitigate IGSCC and (b) performing in-service inspection to monitor IGSCC and its effects on the intended function of BWR piping components within the scope of license renewal. The program is applicable to all BWR piping and piping welds made of austenitic SS and nickel alloy that is 4 inches or larger in nominal diameter and contains reactor coolant at a temperature above 93 degrees Celsius (200 degrees Fahrenheit) during power operation, regardless of code classification. The program also applies to pump casings, valve bodies, and reactor vessel attachments and appurtenances, such as head spray and vent components. NUREG-0313, Rev. 2 and NRC GL 88-01, respectively, describe the technical basis and staff guidance regarding mitigation of IGSCC in BWRs. Attachment A of NRC GL 88-01 delineates the staff-approved positions regarding materials, processes, water chemistry, weld overlay reinforcement, partial replacement, stress improvement of cracked welds, clamping devices, crack characterization and repair criteria, inspection methods and personnel, inspection schedules, sample expansion, leakage detection, and reporting requirements.

2. ***Preventive Actions:*** The BWR Stress Corrosion Cracking Program is a condition monitoring program and has no preventive actions.

   ~~The program delineated in NUREG-0313 and NRC GL 88-01 does not provide specific guidelines for controlling reactor water chemistry to mitigate IGSCC.~~ Maintaining high water purity reduces susceptibility to SCC or IGSCC. . Reactor coolant water chemistry is monitored and maintained in accordance with the Water Chemistry Program.~~The water chemistry program for BWRs relies on monitoring and control of reactor water chemistry based on industry guidelines, such as BWRVIP-190 (Electric Power Research Institute [EPRI] 1016579). BWRVIP-190 has three sets of guidelines: one for primary water, one for condensate and feedwater, and one for control rod drive (CRD) mechanism cooling water.~~ The program description and evaluation and technical basis of monitoring and maintaining reactor water chemistry are addressed ~~through implementation~~in ~~of~~ Section XI.M2, "Water Chemistry."

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

4. ***Detection of Aging Effects:*** The extent, method, and schedule of the inspection and test techniques delineated in NRC GL 88-01 or BWRVIP-75A are designed to maintain structural integrity and ensure that aging effects are discovered and repaired before the loss of intended function of the component. The inspection guidance in approved BWRVIP-75A replaces the extent and schedule of inspection in NRC GL 88-01. The program uses volumetric examinations to detect IGSCC. Inspection can reveal cracking and leakage of coolant. The extent and frequency of inspection recommended by the program are based on the condition of each weld (e.g., whether the weldments were made from IGSCC-resistant material, whether a stress improvement process was applied to a weldment to reduce residual stresses, and how the weld was repaired if it had been cracked).

5. ***Monitoring and Trending:*** The extent and schedule for inspection, in accordance with the recommendations of NRC GL 88 01 or approved BWRVIP-75A guidelines, provide timely detection of cracks and leakage of coolant. Indications of cracking are evaluated and

trended in accordance with the 2004 edition of the American Society of Mechanical Engineers (ASME) Code, Section XI, IWB-3100, against applicable acceptance standard criteria that are specified in the ASME Code, Section XI, IWB-3400 or IWB-3500. Applicable and approved BWRVIP-14A, BWRVIP-59A, BWRVIP-60A, 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.

6. ***Acceptance Criteria:*** Any cracking is evaluated in accordance with ASME Code, Section XI, IWB-3100 by comparing inspection results with the acceptance standards of ASME Code, Section XI, IWB-3400 and ASME Code, Section XI, IWB-3500.
7. ***Corrective Actions***: The guidance for weld overlay repair and stress improvement or replacement is provided in NRC GL 88-01. Corrective action is performed in accordance with IWA-4000. 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:*** Site quality assurance 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 confirmation process and administrative controls.
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:*** Intergranular SCC has occurred in small- and large-diameter BWR piping made of austenitic SS and nickel-base alloys. Cracking has occurred in recirculation, core spray, residual heat removal, CRD return line penetrations, and reactor water cleanup system piping welds (NRC GL 88-01, NRC Information Notices [INs] 82-39, 84-41, and 04-08). The comprehensive program outlined in NRC GL 88-01, NUREG-0313, Rev. 2, and in the staff-approved BWRVIP-75A report addresses mitigating measures for SCC or IGSCC (e.g., susceptible material, significant tensile stress, and an aggressive environment). The GL 88-01 program has been effective in managing IGSCC in BWR reactor coolant pressure-retaining components, and the revision to the GL 88-01 program, according to the staff-approved BWRVIP-75A report, will adequately manage IGSCC degradation.

**References**

10 CFR Part 50, Appendix B, *Quality Assurance Criteria for Nuclear Power Plants*, Office of the Federal Register, National Archives and Records Administration, 2009.

10 CFR 50.55a, *Codes and Standards*, Office of the Federal Register, National Archives and Records Administration, 2009.

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*, The ASME Boiler and Pressure Vessel Code, 2004 edition as approved in 10 CFR 50.55a, The American Society of Mechanical Engineers, New York, NY.

BWRVIP-14A (EPRI 1016569), *BWR Vessel and Internals Project, Evaluation of Crack Growth in BWR Stainless Steel RPV Internals,* Final Safety Evaluation Report by the Office of Nuclear Reactor Regulation, September 2008.

BWRVIP-59A, (EPRI 1014874), *BWR Vessel and Internals Project, Evaluation of Crack Growth in BWR Nickel-Base Austenitic Alloys in RPV Internals*, Final Report by the Office of Nuclear Reactor Regulation, May 2007.

BWRVIP-60A (EPRI 108871), *BWR Vessel and Internals Project, Evaluation of Stress Corrosion Crack Growth in Low Alloy Steel Vessel Materials in the BWR Environment,* Final Safety Evaluation Report by the Office of Nuclear Reactor Regulation, June 2003.

BWRVIP-61 (EPRI 112076), *BWR Vessel and Internals Induction Heating Stress Improvement Effectiveness on Crack Growth in Operating Reactors,* Final Safety Evaluation Report by the Office of Nuclear Reactor Regulation, January 29, 1999.

BWRVIP-62 (EPRI 108705), *BWR Vessel and Internals Project, Technical Basis for Inspection Relief for BWR Internal Components with Hydrogen Injection,* Final Safety Evaluation Report by the Office of Nuclear Reactor Regulation, March 7, 2000.

BWRVIP-75A (EPRI 1012621), *BWR Vessel and Internals Project, Technical Basis for Revisions to Generic Letter 88-01 Inspection Schedules (NUREG-0313),* Final Safety Evaluation Report by the Office of Nuclear Reactor Regulation, October 2005.

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. NUREG-1801, Rev. 2 XI M7-4 April 2010 DRAFT

NRC Information Notice 04-08, *Reactor Coolant Pressure Boundary Leakage Attributable to Propagation of Cracking in Reactor Vessel Nozzle Welds,* U.S. Nuclear Regulatory Commission, April 22, 2004.

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

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

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

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## XI.M9 BWR VESSEL INTERNALS

Comment/Basis:

XI.M9 – 1 General Comment on Program -

The BWR Vessel Internals Program, as defined in Rev. 1 of GALL, described the BWRVIP including inspection and flaw evaluation of vessel internals components in conformance with the guidelines of applicable and staff-approved boiling water reactor vessel and internals project (BWRVIP) documents. The staff proposed changes to this program for GALL, Rev.2, add component aging management guidelines that are outside the guidance of the BWRVIP.

Specifically, the changes proposed by the staff add aging management guidelines for two MEAP combinations;

1) the guidance for CASS internals components exposed to reactor coolant and neutron flux formerly addressed in GALL, Rev. 1, XI.M13, Thermal Aging and Neutron Irradiation Embrittlement of Cast Austenitic Stainless Steel (CASS) ; and

2) guidance for X-750 alloy, and precipitation-hardened (PH) martensitic stainless steel internals components exposed to reactor coolant and neutron flux that (according to the draft GALL master item RP-182) appear to be the subject of a new (but unidentified) ISG.

BWRVIP-234, Thermal Aging and Neutron Embrittlement Evaluation of Cast Austenitic Stainless Steels for BWR Internals, was issued in December 2009, and (presumably) is under staff review. This BWRVIP document would include the GALL, Rev. 1, XI.M13 guidance within the BWRVIP program, but BWRVIP-234 has not been accepted by the staff and is not mentioned in the staff proposed changes to XI.M9. Until, the BWRVIP-234 is accepted, XI.M13 should remain a separate GALL program.

The guidance for X-750 alloy, and precipitation-hardened (PH) martensitic stainless steel internals components exposed to reactor coolant and neutron flux, should be established as a separate GALL program if appropriate, but should not be added to XI.M9 unless it is eventually addressed by the BWRVIP. Suggest adding "in accordance with industry and NRC staff approved guidance" in elements 3, 5 and 6 to allow use of BWRVIP guidance when developed that will eliminate and exception.

XI.M9 – 2 Typo for 'ASME' in element 4.

XI.M9 - 3 In the Program Description, for many BWRs, the actual Mb content is not given on the CMTRs. It was not the practice to sample for Mb, especially in the early years of nuclear construction, unless Mb was specified as an additive. Consequently, there is no way to verify the measured Mb. It is safe to assume that Mb was not added to material unless required. Therefore, CF3, CF3A, CF8, CF8A can be assumed to be low-molybdenum steels without testing. This is consistent with EPRI guidance in TR 100976. See suggested revisions below.

XI.M9-4 The statement about not actually monitoring reduction in fracture toughness that was added to Element 3 is good. I suggest it should also be included in the program description.

XI.M9-5 In SCOPE related to top guides. Every BWR will exceed the fluence threshold prior to the PEO. Most BWRs exceed this threshold in the 4th or 5th fuel cycle. Eliminate the paragraph for those plants that haven't reached the threshold and make the first paragraph the only option.

XI.M9-6 Make the PREVENTIVE ACTIONS section read like all the other condition monitoring programs. See comments on XI.M4.

XI.M9-7 The footnote in Element 3 about using different ASME code versions is different than the footnote in other AMPs. Use the same footnote in all AMPs unless there is a different meaning intended.

**Program Description**

The program includes inspection and flaw evaluations in conformance with the guidelines of applicable and staff-approved boiling water reactor vessel and internals project (BWRVIP) documents to ensure the long-term integrity and safe operation of boiling water reactor (BWR) vessel internal components.

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

In addition, this program provides screening criteria to determine the susceptibility of cast austenitic stainless steels (CASS) components to thermal aging on the basis of casting method, molybdenum content, and percent ferrite, in accordance with the criteria set forth in the May 19, 2000 letter from Christopher Grimes, Nuclear Regulatory Commission (NRC), to Mr. Douglas Walters, Nuclear Energy Institute (NEI). The susceptibility to thermal aging embrittlement of CASS components is determined in terms of casting method, molybdenum content, and ferrite content. For low-molybdenum content steels (SA-351 Grades CF3, CF3A, CF8, CF8A, or other steels with ≤0.5 wt.% molybdenum~~ax.) steels~~, only static-cast steels with >20% ferrite are potentially susceptible to thermal embrittlement. Static-cast low-molybdenum steels with ≤20% ferrite and all centrifugal-cast low-molybdenum steels are not susceptible. For high-molybdenum content steels (SA-351 Grades CF3M, CF3MA, CF8M or other steels with 2.0 to 3.0 wt.% molybdenum)~~steels~~, static-cast steels with >14% ferrite and centrifugal-cast steels with >20% ferrite are potentially susceptible to thermal embrittlement. Static-cast high-molybdenum steels with ≤14% ferrite and centrifugal-cast high-molybdenum steels with ≤20% ferrite are not susceptible. In the susceptibility screening method, ferrite content is calculated by using the Hull's equivalent factors (described in NUREG/CR-4513, Rev. 1) or a method producing an equivalent level of accuracy (±6% deviation between measured and calculated values).

The screening criteria are applicable to all cast stainless steel primary pressure boundary and reactor vessel internal components ~~constructed from SA-351 Grades CF3, CF3A, CF8, CF8A, CF3M, CF3MA, CF8M,~~ with service conditions above 250°C (482°F). The screening criteria for susceptibility to thermal aging embrittlement are not applicable to niobium-containing steels; such steels require evaluation on a case-by-case basis. For "potentially susceptible" components, the program considers synergistic loss of fracture toughness due to neutron embrittlement and thermal aging embrittlement.

The program does not directly monitor for loss of fracture toughness; instead, the impact of loss of fracture toughness on component integrity is indirectly managed by using visual or volumetric examination techniques to monitor for cracking in the components and by applying applicable reduced fracture toughness properties in the flaw evaluations if cracking is detected in the components and is extensive enough to warrant a supplemental flaw growth or flaw tolerance evaluation under the applicable BWRVIP guidelines or ASME Code, Section XI requirements.

This AMP addresses aging degradation of X-750 alloy, and precipitation-hardened (PH) martensitic stainless steel (e.g., 15-5 and 17-4 PH steel) materials and martensitic stainless steel (e.g., 403, 410, 431 steel) that are used in BWR vessel internal components. When exposed to a BWR reactor temperature of 550o F, these materials can experience neutron embrittlement and a decrease in fracture toughness. PH-martensitic stainless steels and martensitic stainless steels are also susceptible to thermal embrittlement. Synergistic effects of thermal and neutron embrittlement can cause failure of these materials in vessel internal components. In addition, X-750 alloy in a BWR environment is susceptible to intergranular stress corrosion cracking (IGSCC).

**Evaluation and Technical Basis**

1. ***Scope of Program:*** The program is focused on managing the effects of cracking due to stress corrosion cracking (SCC), IGSCC, or irradiation-assisted stress corrosion cracking (IASCC). This program also includes loss of toughness due to neutron and thermal embrittlement. The program contains in-service inspection (ISI) to monitor the effects of cracking on the intended function of the components, uses NRC-approved BWRVIP reports as the basis for inspection, evaluation, repair and/or replacement, as needed, and evaluates the susceptibility of CASS, X-750 alloy, precipitation-hardened (PH) martensitic stainless steel (e.g., 15-5 and 17-4 PH steel), and martensitic stainless steel (e.g., 403, 410, 431 steel) components to neutron and/or thermal embrittlement.

   The scope of the program includes the following BWR reactor vessel (RV) and RV internal components as subject to the following NRC-approved applicable BWRVIP guidelines:

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

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

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

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

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

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

*Top guide:* BWRVIP-26A and BWRVIP-183 provide guidelines for inspection and evaluation; BWRVIP-50A provides guidelines for repair design criteria. ~~Additionally, for top guides with neutron fluence exceeding the IASCC threshold (5E20, E>1MeV) prior to the period of extended operation, i~~ Inspect five percent (5%) of the top guide locations using enhanced visual inspection technique, EVT-1 within six years after entering the period of extended operation. An additional 5% of the top guide locations will be inspected within twelve years after entering the period of extended operation.

~~Alternatively, if the neutron fluence for the limiting top guide location is projected to exceed the threshold for IASCC after entering the period of extended operation, inspect 5% of the top guide locations (EVT-1) within six years after the date projected for exceeding the threshold. An additional 5% of the top guide locations will be inspected within twelve years after the date projected for exceeding the threshold.~~

The top guide inspection locations are those that have high neutron fluences exceeding the IASCC threshold. The extent of the examination and its frequency will be based on a ten percent sample of the total population, which includes all grid beam and beam-to-beam crevice slots.

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

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

*Reactor Vessel Internals*: BWRVIP-74A provides guidelines for inspection and evaluation of the aging management and time TLAA for the internals.

*Steam Dryer*: BWRVIP-139 provides guidelines for inspection and evaluation for the steam dryer components.

2. ***Preventive Actions:*** This ~~is a~~ condition monitoring program has no preventive actions.~~;~~ ~~however,~~ ~~m~~Maintaining high water purity reduces susceptibility to cracking due to SCC or IGSCC. Reactor coolant water chemistry is monitored and maintained in accordance with the Water Chemistry Program.~~The water chemistry program for BWRs relies on monitoring and control of reactor water chemistry based on industry guidelines, such as BWRVIP-190 (Electric Power Research Institute [EPRI] 1016579). The BWRVIP-190 has three sets of guidelines: one for primary water, one for condensate and feedwater, and one for control rod drive (CRD) mechanism cooling water.~~ The program description and the evaluation and technical basis of monitoring and maintaining reactor water chemistry are presented in Chapter XI.M2, "Water Chemistry." In addition, the program maintains operating tensile stresses below a threshold limit that precludes IGSCC of X-750 material.

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 (2004 edition[3]).

Loss of fracture toughness is due to thermal and/or neutron embrittlement in CASS materials, can occur due to exposure to neutron fluence of greater than $10^{19}$ n/cm$^2$ (E>1 MeV) or if CASS material is more susceptible to thermal embrittlement due to casting method, molybdenum content, and ferrite content. The program does not directly monitor for loss of fracture toughness that is induced by either thermal aging; neutron irradiation embrittlement, or by void swelling and irradiation growth; instead, the impact of loss of fracture toughness on component integrity is indirectly managed by using visual or volumetric examination techniques to monitor for cracking in the components and by applying applicable reduced fracture toughness properties in the flaw evaluations if cracking is detected in the components and is extensive enough to warrant a supplemental flaw growth or flaw tolerance evaluation under the applicable BWRVIP guidelines or ASME Code, Section XI requirements.

Thermal and/or neutron embrittlement of X-750 alloys, PH-martensitic stainless steels and martensitic stainless steels cannot be identified by typical in-service inspection activities. However, by performing visual or other inspections, applicants can identify cracks that could lead to failure of the embrittled component prior to component failure. Applicants, thus, can prevent the deleterious effects of embrittlement in the PH steels, martensitic stainless steels, and X-750 components by identifying aging degradation (i.e., cracks), implementing early corrective actions, and monitoring and trending age-related degradation in accordance with industry and NRC staff approved guidance.

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

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

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[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.

Thermal and/or neutron embrittlement in susceptible CASS, PH-martensitic steels, martensitic stainless steels, and X-750 components are detected by performing periodic visual inspections. The 10-year ISI program during the renewal period should include a supplemental inspection covering portions of the susceptible components determined to be limiting from the standpoint of thermal aging susceptibility, neutron fluence, and cracking susceptibility (i.e., applied stress, operating temperature, and environmental conditions). The inspection technique is capable of detecting the critical flaw size with adequate margin. The critical flaw size is determined based on the service loading condition and service-degraded material properties. One example of a supplemental examination is EVT-1 examination of AMSE Code, Section XI, IWA-2210. The initial inspection is performed either prior to or within 5 years after entering the period of extended operation. If cracking is detected after the initial inspection, the frequency of re-inspection should be justified by the applicant based on assessed fracture toughness properties. The sample size is 100% of the accessible component population, excluding components that may be in compression during normal operations.

5. ***Monitoring and Trending:*** Inspections are scheduled in accordance with the applicable and approved BWRVIP guidelines provide timely detection of cracks. Each BWRVIP guideline recommends baseline inspections that are used as part of data collection towards trending. The BWRVIP guidelines provide recommendations for expanding the sample scope and re-inspecting the components if flaws are detected. Any indication detected is evaluated in accordance with ASME Code, Section XI or the applicable BWRVIP guidelines. BWRVIP-14A, BWRVIP-59A, and BWRVIP-60A documents provide additional guidelines for evaluation of crack growth in stainless steels (SSs), nickel alloys, and low-alloy steels, respectively.

   A fracture toughness value of 255 kJ/m2 (1,450 in.-lb/in.2) at a crack depth of 2.5 mm (0.1 in.) is used to differentiate between CASS materials that are susceptible to thermal aging embrittlement and those that are not. Extensive research data indicate that for non-susceptible CASS materials, the saturated lower-bound fracture toughness is greater than 255 kJ/m2 (NUREG/CR-4513, Rev. 1).

   Inspections scheduled in accordance with ASME Code, Section XI, IWB-2400 and reliable examination methods provide timely detection of cracks. The fracture toughness of PH-martensitic steels, martensitic stainless steels, and X-750 alloys susceptible to thermal and/or neutron embrittlement need to be assessed on a case-by-case basis <u>or be in accordance with industry and NRC staff approved guidance</u>.

6. ***Acceptance Criteria:*** Acceptance criteria are given in the applicable BWRVIP documents or ASME Code, Section XI. Flaws detected in CASS components are evaluated in accordance with the applicable procedures of ASME Code, Section XI, IWB-3500. Flaw tolerance evaluation for components with ferrite content up to 25% is performed according to the principles associated with ASME Code, Section XI, IWB-3640 procedures for submerged arc welds (SAW), disregarding the Code restriction of 20% ferrite in ASME Code, Section XI, 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.

Acceptance criteria for the assessment of PH-martensitic steels, martensitic stainless steels, and X-750 alloys susceptible to thermal aging and/or neutron embrittlement are assessed on a case-by-case basis or are in accordance with industry and NRC staff approved guidance.

7. ***Corrective Actions:*** Repair and replacement procedures are equivalent to those requirements in ASME Code Section XI. Repair and replacement is performed in conformance with the applicable and approved BWRVIP guidelines listed above. For top guides where cracking is observed, sample size and inspection frequencies ar increased. As discussed in the appendix to this report, the staff finds that licensee implementation of the corrective action guidelines in the staff-approved BWRVIP reports will provide an acceptable level of quality accordance with 10 CFR Part 50, Appendix B.

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

9. ***Administrative Controls:*** 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:*** There is documentation of cracking in both the circumferential and axial core shroud welds. Extensive cracking of circumferential core shroud welds has been documented in NRC Generic Letter 94-03 and extensive cracking in vertical core shroud welds has been documented in NRC Information Notice 97-17. It has affected shrouds fabricated from Type 304 and Type 304L SS, which is generally considered to be more resistant to SCC. Weld regions are most susceptible to SCC, although it is not clear whether this is due to sensitization and/or impurities associated with the welds or the high residual stresses in the weld regions. This experience is reviewed in NRC GL 94-03 and NUREG-1544; some experiences with visual inspections are discussed in NRC IN 94-42.

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

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

    Instances of cracking have occurred in the jet pump assembly (NRC Bulletin 80-07), hold-down beam (NRC IN 93-101), and jet pump riser pipe elbows (NRC IN 97-02). Cracking of CRD dry tubes has been observed at 14 or more BWRs. The cracking is intergranular and has been observed in dry tubes without apparent sensitization, suggesting that IASCC may also play a role in the cracking.

    Two CRDM lead screw male couplings were fractured in a pressurized-water reactor (PWR), designed by Babcock and Wilcox (B&W), at Oconee Nuclear Station (ONS), Unit 3. The fracture was due to thermal embrittlement of 17-4 PH material (NRC IN 2007-02). While this occurred at a PWR, it also needs to be considered by BWRs.

IGSCC in the X-750 materials of a tie rod coupling was observed in a domestic plant.

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

**References**

10 CFR Part 50, Appendix B, *Quality Assurance Criteria for Nuclear Power Plants*, Office of the Federal Register, National Archives and Records Administration, 2009.

10 CFR 50.55a, *Codes and Standards*, Office of the Federal Register, National Archives and Records Administration, 2009.

ASME Section XI, *Rules for Inservice Inspection of Nuclear Power Plant Components*, The ASME Boiler and Pressure Vessel Code, 2004 edition as approved in 10 CFR 50.55a, The American Society of Mechanical Engineers, New York, NY. April 2010 DRAFT XI M9-7 NUREG-1801, Rev. 2

## XI.M11B CRACKING OF NICKEL-ALLOY COMPONENTS AND LOSS OF MATERIAL DUE TO BORIC ACID-INDUCED CORROSION IN REACTOR COOLANT PRESSURE BOUNDARY COMPONENTS (PWRs ONLY)

Comment/Basis:

XI.M11B-1 Removed discussion of the water chemistry guidelines and just referred to the GALL water chemistry program in Program Description.

XI.M11B-2 Change element 2 to match element 2 in other condition monitoring programs. In particular delete the reference to the EPRI document and just refer to XI.M2.

XI.M11B-3 Element 3 needs revised because PWSCC and BAC never apply to the same component. PWSCC applies to nickel alloy components and BAC applies to steel components.

XI.M11B-4 The discussion of Water Chemistry in Element 10 sounds like this program includes maintaining water chemistry. That discussion should be altered to make it clear that XI.M2 maintains water chemistry.

## XI.M11B CRACKING OF NICKEL-ALLOY COMPONENTS AND LOSS OF MATERIAL DUE TO BORIC ACID-INDUCED CORROSION IN REACTOR COOLANT PRESSURE BOUNDARY COMPONENTS (PWRs ONLY)

### Program Description

This program replaces AMPs XI.M11, "Nickel-Alloy Nozzles and Penetrations" and XI.M11A, "Nickel-Alloy Penetration Nozzles Welded to the Upper Reactor Vessel Closure Heads of Pressurized Water Reactors." It addresses the issue of cracking of nickel-alloy components and loss of material due to boric acid-induced corrosion in susceptible, safety-related components in the vicinity of nickel-alloy reactor coolant pressure boundary components. A final rule (September, 2008) updating 10 CFR 50.55a requires the following American Society of Mechanical Engineer (ASME) Boiler and Pressure Vessel (B&PV) Code Cases: (a) N-722, "Additional Examinations for PWR Pressure Retaining Welds in Class 1 Components Fabricated with Alloy 600/82/182 Materials, Section XI, Division 1" to establish long-term inspection requirements for the pressurized water reactor (PWR) vessel, steam generator, pressurizer components and piping if they contain the primary water stress corrosion cracking (PWSCC) susceptible materials designated alloys 600/82/182; and (b) N-729-1, "Alternative Examination Requirements for PWR Reactor Vessel Upper Heads With Nozzles Having Pressure-Retaining Partial-Penetration Welds, Section XI, Division 1" to establish new requirements for the long-term inspection of reactor pressure vessel upper heads.

In addition , dissimilar metal welds need additional examinations to provide reasonable assurance of structural integrity. The U.S. Nuclear Regulatory Commission (NRC) issued Regulatory Information Summary (RIS) 2008-25, "Regulatory Approach for Primary Water

Stress Corrosion Cracking (PWSCC) of Dissimilar Metal Butt Welds in Pressurized Water Reactor Primary Coolant System Piping" (October 2008) which stated the regulatory approach for addressing PWSCC of dissimilar metal butt welds. The RIS documents the NRC's approach for ensuring the integrity of primary coolant system piping containing dissimilar metal butt welds in PWRs and, in conjunction with the mandated inspections of ASME Code Case N-722, ensures that augmented in-service inspections (ISI) of all nickel-based alloy components and welds in the reactor coolant system (RCS) continue to perform their intended functions.

As stated in this RIS, the NRC has found that MRP-139, "Primary System Piping Butt Weld Inspection and Evaluation Guideline" (2005), and MRP interim guidance letters provide adequate protection of public health and safety for addressing PWSCC in dissimilar metal butt welds pending the incorporation of ASME Code Case N-770 containing comprehensive inspection requirements into 10 CFR 50.55a. It is the intention of the NRC to replace MRP-139 by incorporating the requirements of ASME Code Case N-770 into 10 CFR 50.55a.

The impacts of boric acid leakage from non-nickel alloy reactor coolant pressure boundary components is addressed in Chapter XI.M10, "Boric Acid Corrosion." The water chemistry program for PWRs relies on monitoring and control of reactor water chemistry ~~based on industry guidelines such as EPRI 1014986 (PWR Primary Water Chemistry Guidelines-Revision 6) and EPRI 1016555 (PWR Secondary Water Chemistry Guidelines-Revision 7)~~ as described in Chapter XI.M2, "Water Chemistry."

**Evaluation and Technical Basis**

1. ***Scope of Program:*** The program is focused on managing the effects of cracking due to PWSCC of all susceptible nickel alloy-based components of the reactor coolant pressure boundary (including nickel-alloy welds). The program also manages the loss of material due to boric acid corrosion in susceptible components in the vicinity of nickel-alloy components. These components could include, but are not limited to, the reactor vessel components (reactor pressure vessel upper head), steam generator components (nozzle-to-pipe connections, instrument connections, and drain tube penetrations), pressurizer components (nozzle-to-pipe connections, instrument connections, and heater penetrations), and reactor coolant system piping (instrument connections and full penetration welds).

2. ***Preventive Actions:*** This program is a condition monitoring program and does not include preventive or mitigative measures.

   ~~However, m~~Maintaining high water purity reduces susceptibility to PWSCC. Reactor coolant water chemistry is monitored and maintained in accordance with the ~~guidelines in EPRI 1014986, Revision 6.~~ The program description and the evaluation and technical basis of monitoring and maintaining reactor water chemistry are presented in Chapter XI.M2, "Water Chemistry."

   ~~At the discretion of the applicant, preventive actions to mitigate PWSCC may be addressed by various measures (e.g., weld overlays, replacement of components with more PWSCC-resistant materials, etc.).~~

3. ***Parameters Monitored/Inspected:*** This is a condition monitoring program that monitors cracking/PWSCC for nickel-alloy penetrations and loss of material by boric acid corrosion for nearby steel~~each~~ components. Reactor coolant pressure boundary cracking and leakage are monitored by the applicant's in-service inspection program in accordance with 10 CFR 50.55a and industry guidelines (e.g., MRP-139). Boric acid deposits, borated water

leakage, or the presence of moisture that could lead to the identification of cracking or loss of material can be monitored through visual examination.

4. ***Detection of Aging Effects:*** The program detects the effect of aging by various methods including non-destructive examination techniques. Reactor coolant pressure boundary leakage can be monitored through the use of radiation air monitoring and other general area radiation monitoring, and technical specifications for reactor coolant pressure boundary leakage. The specific types of non-destructive examinations are dependent on the component's susceptibility to PWSCC and its accessibility to inspection. Inspection methods, schedules, and frequencies for the susceptible components are implemented in accordance with 10 CFR 50.55a and industry guidelines (e.g., MRP-139).

5. ***Monitoring and Trending:*** Reactor coolant pressure boundary leakage is calculated and trended on a routine basis in accordance with technical specification to detect changes in the leakage rates. Flaw evaluation through 10 CFR 50.55a is a means to monitor cracking.

6. ***Acceptance Criteria:*** Acceptance criteria for all indications of cracking and loss of material due to boric acid-induced corrosion are defined in 10 CFR 50.55a and industry guidelines (e.g., MRP-139).

7. ***Corrective Actions:*** Relevant flaw indications of susceptible components within the scope of this program found to be unacceptable for further services are corrected through implementation of appropriate repair or replacement as dictated by 10 CFR 50.55a and industry guidelines (e.g., MRP-139). In addition, detection of leakage or evidence of cracking in susceptible components within the scope of this program require scope expansion of current inspection and increased inspection frequencies of some components as required by 10 CFR 50.55a and industry guidelines (e.g., MRP-139).

   Repair and replacement procedures and activities must either comply with ASME Section XI, as incorporated in 10 CFR 50.55a or conform to applicable ASME Code Cases that have been endorsed in 10 CFR 50.55a by referencing the latest version of NRC Regulatory Guide 1.147.

   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:*** Site quality assurance procedures and review and approval processes 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 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:*** This new program addresses reviews of related operating experience, including plant-specific information, generic industry findings, and international data. Within the current regulatory requirements, as necessary, the applicant maintains a record of operating experience through the required update of the facility's inservice inspection program in accordance with 10 CFR 50.55a. Additionally, the applicant follows mandated industry guidelines developed to address operating experience in accordance with NEI-03-08, "Guideline for the Management of Materials Issues."

Cracking of Alloy 600 has occurred in domestic and foreign PWRs (NRC Information Notice [IN] 90-10). Furthermore, ingress of demineralizer resins also has occurred in operating plants (NRC IN 96-11). The Water Chemistry P ~~p~~rogram (XI.M2) manages the effects of such excursions through ~~relies upon~~ monitoring and control of primary water chemistry ~~to manage the effects of such excursions~~. NRC GL 97-01 is effective in managing the effect of PWSCC. PWSCC also is occurring in the vessel head penetration (VHP) nozzle of U.S. PWRs as described in NRC Bulletins 2001-01, 2002-01 and 2002-02.

**References**


10 CFR Part 50, Appendix B, *Quality Assurance Criteria for Nuclear Power Plants*, Office of the Federal Register, National Archives and Records Administration, 2009.

10 CFR Part 50.55a, *Codes and Standards*, Office of the Federal Register, National Archives and Records Administration, 2009.

ASME Code Case N-722, *Additional Examinations for PWR Pressure Retaining Welds in Class 1 Components Fabricated with Alloy 600/82/182 Materials*, July 5, 2005.

ASME Code Case N-729-1, *Alternative Examination Requirements for PWR Reactor Vessel Upper Heads with Nozzles Having Pressure-Retaining Partial-Penetration Welds*, March 28, 2006.

ASME Code Case N-770, *Alternative Examination Requirements and Acceptance Standards for Class 1 PWR Piping and Vessel Nozzle Butt Welds Fabricated with UNS N06082 or UNS W86182 Weld Filler Material With or Without Application of Listed Mitigation Activities*, January 26, 2009.NUREG-1801, Rev. 2

EPRI 1014986, *PWR Primary Water Chemistry Guidelines,* Revision 6, Volumes 1 and 2, Electric Power Research Institute, Palo Alto, CA, December 2007

MRP-139, Revision 1, *Primary System Piping Butt Weld Inspection and Evaluation Guideline*, Materials Reliability Program, December 16, 2008.

NEI-03-08, *Guideline for the Management of Materials Issues,* Nuclear Energy Institute, May 2003.

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

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

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

NRC Inspection Manual, Inspection Procedure 71111.08, *Inservice Inspection Activities*, March 23, 2009.

NRC Inspection Manual, Temporary Instruction 2515/172, *Reactor Coolant System Dissimilar Metal Butt Welds*, February 21, 2008.

NRC Regulatory Guide 1.147, Revision 15, *Inservice Inspection Code Case Acceptability*, ASME Section XI, Division 1, U.S. Nuclear Regulatory Commission, January 2004.

NRC Regulatory Information Summary 2008-25, *Regulatory Approach for Primary Water Stress Corrosion Cracking of Dissimilar Metal Butt Welds in Pressurized Water Reactor Primary Coolant System Piping*, U.S. Nuclear Regulatory Commission, October 22, 2008.

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Bulletin 2002-02, Reactor Pressure Vessel Head and Vessel Head Penetration Nozzle Inspection Programs, U.S. Nuclear Regulatory Commission, August 9, 2002.

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

Comment/Basis :

XI.M12-1 See the comments on XI.M9. CF3, CF3A, CF8, CF8A should be considered low-molybdenum steels.

XI.M12-2 PREVENTIVE ACTIONS should be the same as other condition monitoring programs.

XI.M12-3 *Suggest that the paragraph added to M09 about detection of reduction in fracture* toughness also be added to Element 3 of this program.

## 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) piping components except for pump casings and valve bodies. This aging management program (AMP) includes determination of the susceptibility of CASS components to thermal aging embrittlement based on casting method, molybdenum content, and percent ferrite. For "potentially susceptible" components, as defined below, aging management is accomplished through either (a) qualified visual inspections, such as enhanced visual examination (EVT-1); (b) a qualified ultrasonic testing (UT) methodology; or (c) a component-specific flaw tolerance evaluation in accordance with the ASME Code, Section XI, 2004 edition[4]. Additional inspection or evaluations to demonstrate that the material has adequate fracture toughness are not required for components that are not susceptible to thermal aging embrittlement.

For pump casings and valve bodies, based on the results of the assessment documented in the letter dated May 19, 2000, from Christopher Grimes, Nuclear Regulatory Commission (NRC), to Douglas Walters, Nuclear Energy Institute (NEI) (May 19, 2000 NRC letter), *screening for* susceptibility to thermal aging embrittlement is not required. The existing ASME Code, 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.

Aging management of CASS reactor internal components of pressurized water reactors (PWRs) are discussed in AMP XI.M16 and for boiling water reactor (BWR) CASS reactor internal components in AMP XI.M9.

### Evaluation and Technical Basis

[4] Refer to the GALL Report, Chapter I, for applicability of other editions of ASME Code, Section XI.

1. ***Scope of Program:*** This program manages loss of fracture toughness in potentially susceptible ASME Code Class 1 piping components made from CASS. The program includes screening criteria to determine which CASS components are potentially susceptible to thermal aging embrittlement and require augmented inspection. The screening criteria are applicable to all primary pressure boundary components constructed from cast austenitic stainless steel~~SA-351 Grades CF3, CF3A, CF8, CF8A, CF3M, CF3MA, and CF8M~~ with service conditions above 250°C (482°F). The screening criteria for susceptibility to thermal aging embrittlement are not applicable to niobium-containing steels; such steels require evaluation on a case-by-case basis.

   Based on the criteria set forth in the May 19, 2000, NRC letter, the susceptibility to thermal aging embrittlement of CASS materials is determined in terms of casting method, molybdenum content, and ferrite content. For low-molybdenum content steels (SA-351 Grades CF3, CF3A, CF8, CF8A or other steels with ≤ 0.5 weight percent [wt.%] mb~~aximum~~) ~~steels~~, only static-cast steels with greater than 20 percent ferrite are potentially susceptible to thermal embrittlement. Static-cast low-molybdenum steels with less than or equal to 20 percent ferrite and all centrifugal-cast low-molybdenum steels are not susceptible. For high-molybdenum content steels (SA-351 Grades CF3M, CF3MA, and CF8M or other steels with 2.0 to 3.0 wt.% mb) ~~steels~~, static-cast steels with greater than 14 percent ferrite and centrifugal-cast steels with greater than 20 percent ferrite are potentially susceptible to thermal embrittlement. Static-cast high-molybdenum steels with less than or equal to 14 percent ferrite and centrifugal-cast high-molybdenum steels with less than or equal to 20 percent ferrite are not susceptible. In the susceptibility screening method, ferrite content is calculated by using the Hull's equivalent factors (described in NUREG/CR-4513, Rev. 1) or a method producing an equivalent level of accuracy (plus or minus 6 percent deviation between measured and calculated values). A fracture toughness value of 255 kilojoules per square meter (kJ/m$^2$) (1,450 inches-pounds per square inch) at a crack depth of 2.5 millimeters (0.1 inch) is used to differentiate between CASS materials that are not susceptible and those that are potentially susceptible to thermal aging embrittlement. Extensive research data indicate that for CASS materials not susceptible to thermal aging embrittlement, the saturated lower-bound fracture toughness is greater than 255 kJ/m$^2$ (NUREG/CR-4513, Rev. 1).

   For pump casings and valve bodies, screening for susceptibility to thermal aging embrittlement is not needed (and thus there are no aging management review line items). For all pump casings and valve bodies greater than a nominal pipe size (NPS) of 4 inches, the existing ASME Code, Section XI inspection requirements, including the alternative requirements of ASME Code Case N-481 for pump casings, are adequate. ASME Code, Section XI, Subsection IWB requires only surface examination of valve bodies less than a NPS of 4 inches. For these valve bodies less than a NPS of 4 inches, the adequacy of inservice inspection (ISI) according to ASME Code, Section XI has been demonstrated by an NRC-performed bounding integrity analysis (Reference letter from Christopher Grimes dated May 19, 2000).

2. ***Preventive Actions:*** This~~e~~ program is a condition monitoring program~~consists of evaluation and inspection~~ and does not~~provides no guidance on methods to~~ mitigate thermal aging embrittlement.

3. ***Parameters Monitored/Inspected:*** The AMP monitors the effects of loss of fracture toughness on the intended function of the component by identifying the CASS materials that are susceptible to thermal aging embrittlement. For potentially susceptible materials, the

program consists of either (a) qualified visual inspections, such as enhanced visual examination (EVT-1); (b) a qualified UT methodology; or (c) a plant- or component-specific flaw tolerance evaluation.

The program does not directly monitor for loss of fracture toughness that is induced by thermal aging; instead, the impact of loss of fracture toughness on component integrity is indirectly managed by using visual or volumetric examination techniques to monitor for cracking in the components and by applying applicable reduced fracture toughness properties in the flaw evaluations if cracking is detected in the components and is extensive enough to warrant a supplemental flaw growth or flaw tolerance evaluation under the applicable BWRVIP guidelines or ASME Code, Section XI requirements.

4. ***Detection of Aging Effects:*** For pump casings, valve bodies, and other "not susceptible" CASS piping components, no additional inspection or evaluations are needed to demonstrate that the material has adequate fracture toughness.

   For "potentially susceptible" piping components, the CASS AMP provides for qualified inspections of the base metal, such as enhanced visual examination (EVT-1) or a qualified UT methodology, with the scope of the inspection covering the portions determined to be limiting from the standpoint of applied stress, operating time, and environmental considerations. Examination methods that meet the criteria of the ASME Code, Section XI, Appendix VIII are acceptable. Alternatively, a plant-specific or component-specific flaw tolerance evaluation, using specific geometry, stress information, material properties, and ASME Code, Section XI can be used to demonstrate that the thermally-embrittled material has adequate toughness. Current UT methodology cannot detect and size cracks; thus, EVT-1 is used until qualified UT methodology for CASS can be established. A description of EVT-1 is found in Boiling Water Reactor Vessel and Internals Project (BWRVIP)-03 (revision 6) and Materials Reliability Program (MRP)-228.

5. ***Monitoring and Trending:*** Inspection schedules in accordance with ASME Code, Section XI, IWB-2400 or IWC-2400, reliable examination methods, and qualified inspection personnel provide timely and reliable detection of cracks. If flaws are detected, the period of acceptability is determined from analysis of the flaw, depending on the crack growth rate and mechanism.

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

7. ***Corrective Actions:*** Repair and replacement are performed in accordance with ASME Code, Section XI, IWA-4000. 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:*** Site quality assurance 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 confirmation process and administrative controls.

9. ***Administrative Controls:*** The administrative controls for this program provide for a formal review and approval of corrective actions. The administrative controls for this program are implemented through the site's QA program in accordance with the requirements of 10 CFR Part 50, Appendix B.

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

**References**

10 CFR Part 50, Appendix B, *Quality Assurance Criteria for Nuclear Power Plants*, Office of the Federal Register, National Archives and Records Administration, 2009.

10 CFR Part 50.55a, *Codes and Standards*, Office of the Federal Register, National Archives and Records Administration, 2009.

ASME Section XI, *Rules for Inservice Inspection of Nuclear Power Plant Components*, The ASME Boiler and Pressure Vessel Code, 2004 edition as approved in 10 CFR 50.55a, The American Society of Mechanical Engineers, New York, NY.

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

Letter from Christopher I. Grimes, U.S. Nuclear Regulatory Commission, License Renewal and Standardization Branch, to Douglas J. Walters, Nuclear Energy Institute, License Renewal Issue No. 98-0030, *Thermal Aging Embrittlement of Cast Stainless Steel Components*, May 19, 2000. (ADAMS Accession No. ML003717179)

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

BWRVIP-03, Rev. 6, *BWR Vessel and Internals Project: Reactor Pressure Vessel and Internals Examination Guidelines* (EPRI TR-105696).

Letter from Mark J. Maxin, to Rick Libra (BWRVIP Chairman), Safety Evaluation for Electric Power Research Institute (EPRI) Boiling Water Reactor Vessel and Internals project (BWRVIP) Report TR-105696-R6 (BWRVIP-03), Revision 6, BWR Vessel and Internals Examination Guidelines (TAC No MC2293)," June 30, 2008 (ADAMS Accession No ML081500814)

MRP-228, *Materials Reliability Program: Inspection Standard for PWR Internals,* 2009.

ASME Code Case N-481, Alternative Examination Requirements for Cast Austenitic Pump Casings, Section XI, Division 1

## XI.M16A PWR Vessel Internals

Comment/Basis:

XI.M16A – 1 The program description seems to be overly detailed and descriptive for a general description when it could more easily just refer to MRP-227 for most of the items. Some items are also repeated in the 10 elements where this detail is appropriate.

XI.M16A – 2 In element 2 preventive actions there is a discussion of applicability limitations for MRP-227. These should not be in preventive actions but in scope section.

XI.M16A – 3 In element 1, 3 and 4 there are sentences that state there is an administrative action item for the applicant to fill in the type of plant and vendor. This is not used for any other GALL programs and not sure why this needs to be written this way. The applicant will develop their program based on the GALL program requirements depending on the make and vintage of his plant.

## XI.M16A PWR VESSEL INTERNALS

**Program Description**

This program relies on implementation of the Electric Power Research Institute (EPRI) Report No. 1016596 (MRP-227) and EPRI Report No. 1016609 (MRP-228) to manage the aging effects on the reactor vessel internal (RVI) components.

~~This program includes the following:~~

~~• Examinations and other inspections, and comparison of these data with examination acceptance criteria, as defined in MRP-227, Revision 0 and MRP-228, Revision 0~~

~~• Disposition of indications that exceed examination acceptance criteria by entering them into the applicant's Corrective Action Program; may include evaluation for continued service until the next examination~~

~~• Monitoring and control of reactor primary coolant water chemistry for pressurized water reactors (PWRs), which relies on monitoring and control of reactor water chemistry based on industry guidelines such as EPRI 1014986 (PWR Primary Water Chemistry Guidelines-Revision 6) and EPRI 1016555 (PWR Secondary Water Chemistry Guidelines-Revision 7)~~

This program is used to manage the effects of age-related degradation mechanisms that are applicable in general to the PWR RVI components at the facility. These aging effects include (a) various forms of cracking, including stress corrosion cracking (SCC), primary water stress corrosion cracking (PWSCC), irradiation-assisted stress corrosion cracking (IASCC), or cracking due to fatigue/cyclical loading; (b) loss of material induced by wear; (c) loss of fracture toughness due to either thermal aging or neutron irradiation embrittlement; (d) dimensional changes and potential loss of fracture toughness due to void swelling and irradiation growth; and (e) loss of preload due to thermal and irradiation-enhanced stress relaxation or creep. ~~The MRP-227 relies on preventive measures, such as fuel loading management and primary water chemistry control, to limit the degradation.~~

The program applies the guidance in MRP-227 for inspecting, evaluating, and, if applicable, dispositioning non-conforming RVI components at the facility. The program conforms to the definition of a sampling-based condition monitoring program, as defined by the Branch Technical Position RSLB-1, with periodic examinations and other inspections of highly-affected internals locations. These examinations provide reasonable assurance that the effects of age-related degradation mechanisms will be managed during the period of extended operation. The

program includes expanding periodic examinations and other inspections if the extent of the degradation effects exceeds the expected levels.
The MRP-227 guidance basis for selecting RVI components for inclusion in the inspection sample is based on a four-step ranking process.:

~~• Screening of reactor internals for all three Babcock & Wilcox (B&W), Combustion Engineering (CE), and Westinghouse designs, considering material properties (e.g., chemical composition) and operational conditions (e.g., neutron fluence exposure, temperature history, and representative stress levels) in order to determine the susceptibility or non-susceptibility of PWR internals to the postulated aging mechanisms~~

~~• Further categorization of these reactor internals, based on the screening results and the likelihood/severity of safety consequences, into categories (for each degradation effect) ranging from insignificant effects (Category A) to potentially moderately significant effects (Category B) to potentially significant effects (Category C)~~

~~• Functionality assessment of components and assemblies of components based on representative plant designs using irradiated and aged material properties to determine the effects of the degradation mechanisms on functionality~~

~~• Aging management strategy development combining the results of functionality assessment with component accessibility, operating experience, existing evaluations, and prior examination results to determine the appropriate methodologies for maintaining the long-term functions of PWR internals safely~~

The result of this four-step sample selection process is a set of Primary Internals Component locations for each of the three plant designs that are expected to show the leading indications of the degradation effects, with another set of Expansion Internals Component locations that are specified to expand the sample should the indications be more severe than anticipated. The degradation effects in a third set of internals locations are deemed to be adequately managed by Existing Programs, such as ASME Code, Section XI (which is endorsed by reference in 10 CFR 50.55a), Examination Category B-N-3 examinations of core support structures. A fourth set of internals locations are deemed to require no additional measures. As a result, the program typically identifies 5 to 15 percent of the RVI locations as Primary Component locations for inspections, with another 7 to 10 percent of the RVI locations to be inspected as Expansion Components, as warranted by the evaluation of the inspection results. Another 5 to 15 percent of the internals locations are covered by Existing Programs, with the remainder requiring no additional measures. Thus, this process uses appropriate component functionality criteria, age-related degradation susceptibility criteria, and failure consequence criteria to identify the components that will be inspected under the program in a manner that conforms to the sampling criteria for sampling-based condition monitoring programs in Section A.1.2.3.4 of NRC Branch Position RLSB-1, and thus, the sample selection process is adequate to assure safety function integrity of the subject safety related PWR reactor internal components.

~~The methodology and guidance in MRP-227 includes information on component description and function (Section 3); requirements for methods, extent, and frequency of the examinations (Section 4); examination acceptance criteria and requirements for expanding the scope of the examinations as needed (Section 5); information on acceptable methods for evaluating the structural integrity significance of flaws detected during these examinations that exceed examination acceptance criteria (Section 6); and general information on component repair and replacement procedures (Section 6). The methodology and guidance also contains provisions for reporting to the EPRI MRP by the individual utilities on the results of the examinations, with the intent that the sampling program extends beyond an individual plant to include all other~~

~~PWRs. In this way, the combined results from many sets of internals examinations are used to determine the need for program adjustments.~~
The program's use of visual examination methods in MRP-227 for detection of relevant conditions (and the absence of relevant conditions as a visual examination acceptance criterion) is consistent with the ASME Code, Section XI rules for visual examination. However, the program's adoption of the MRP-227 guidance for visual examinations goes beyond the ASME Code, Section XI visual examination criteria because additional guidance is incorporated into MRP-227 to clarify how the particular visual examination methods will be used to detect relevant conditions and describes in more detail how the visual techniques relate to the specific RVI components and how to detect their applicable age-related degradation effects.

The technical basis for detecting relevant conditions using volumetric ultrasonic testing (UT) inspection techniques can be found in MRP-228, where the review of existing bolting UT examination technical justifications has demonstrated the indication detection capability of at least two vendors, and where vendor technical justification is a requirement prior to any additional bolting examinations. Specifically, the capability of program's UT volumetric methods to detect loss of integrity of PWR internals bolts, pins, and fasteners, such as baffle-former bolting in B&W and Westinghouse units, has been well demonstrated by operating experience. In addition, the program's adoption of the MRP-227 guidance and process incorporates the UT criteria in MRP-228, which calls for the technical justifications that are needed for volumetric examination method demonstrations, required by the ASME Code, Section V.

~~For some components, the MRP-227 methodology specifies a focused visual (VT-3) examination, similar to the current ASME Code, Section XI, Examination Category B-N-3 examinations, in order to determine the general mechanical and structural condition of the internals by (a) verifying parameters, such as clearances, settings, and physical displacements; and (b) detecting discontinuities and imperfections, such as loss of integrity at bolted or welded connections, loose or missing parts, debris, corrosion, wear, or erosion. In some cases, VT-3 visual methods are used for the detection of surface cracking when the component material has been shown to be tolerant of easily detected large flaws. Otherwise more rigorous detection of cracking is required, and the PWR internals will be examined by visual (VT-1) examination, in order to detect discontinuities and imperfections, such as cracks, corrosion, wear, or erosion, on the surfaces of components. In some cases, where even more stringent examinations are required, enhanced visual (EVT-1) examinations or ultrasonic methods of volumetric inspection, are specified for certain selected components and locations.~~
The program also includes future industry operating experience as incorporated in periodic revisions to MRP-227. The program thus ensures the long-term integrity and safe operation of reactor internals in all commercial operating U.S. PWR nuclear power plants.

**Evaluation and Technical Basis**

**1. Scope of Program:** The scope of the program includes all RVI components at the [as an administrative action item for the AMP, the applicant to fill in the name of the applicant's nuclear facility, including applicable units], which [is/are] built to a [applicant to fill in Westinghouse, CE, or B&W, as applicable] NSSS design. The scope of the program applies the methodology and guidance in the most recently NRC-endorsed version of MRP-227, which provides augmented inspection and flaw evaluation methodology for assuring the functional integrity of safety-related internals in commercial operating U.S. PWR nuclear power plants designed by B&W, CE, and Westinghouse. The scope of components considered for inspection under MRP-227 guidance includes core support structures (typically denoted as Examination Category B-N-3 by the ASME Code, Section XI), those RVI components that serve a intended license renewal safety function pursuant to criteria in 10 CFR 54.4(a)(1), and other RVI components whose failure may impact the ability of a component with an intended license renewal safety function to achieve its intended safety related objective (10 CFR 54.4(a)(2)). The scope of the program does not include consumable items such as fuel assemblies, reactivity control assemblies, and nuclear instrumentation because these components are not typically within the scope of the components that are required to be subject to an aging management review (AMR), as defined by the criteria set in 10 CFR 54.21(a)(1). The scope of the program also does not include welded attachments to the internal surface of the reactor vessel because these components are considered to be ASME Code Class 1 appurtenances to the reactor vessel and are adequately managed in accordance with an applicant's AMP that corresponds to GALL AMP XI.M1, "ASME Code, Section XI Inservice Inspection, Subsections IWB, IWC, and IWD."

The scope of the program includes the response bases to applicable license renewal applicant action items (LRAAIs) on the MRP-227 methodology, and any additional programs, actions, or activities that are discussed in these LRAAI responses and credited for aging management of the applicant's RVI components. The LRAAIs are identified in the staff's safety evaluation on MRP-227 and include applicable action items on meeting those assumptions that formed the basis of the MRP's augmented inspection and flaw evaluation methodology (as discussed in Section 2.4 of MRP-227), and NSSS vendor-specific or plant-specific LRAAIs as well. The responses to the LRAAIs on MRP-227 are provided in Appendix C of the LRA.

The guidance in MRP-227 specifies applicability limitations to base-loaded plants and the fuel loading management assumptions upon which the functionality analyses were based. These limitations and assumptions require a determination of applicability by the applicant for each reactor and are covered in Section 2.4 of MRP-227.

2. ***Preventive Actions:*** ~~The guidance in MRP-227 does not specify any preventive actions other than the applicability limitations to base-loaded plants and the fuel loading management assumptions upon which the functionality analyses were based. These limitations and assumptions require a determination of applicability by the applicant for each reactor and are covered in Section 2.4 of MRP-227.~~

~~In addition,~~ the guidance in MRP-227 relies on PWR water chemistry control to prevent or mitigate the effects of aging effects that can be induced by corrosive aging mechanisms (e.g., loss of material induced by general, pitting corrosion, crevice corrosion, or stress corrosion cracking or any of its forms [SCC, PWSCC, or IASCC]). Therefore, an important adjunct to the aging management methodologies described by the guidance in MRP-227 is PWR water chemistry control. The water chemistry program for PWRs relies on monitoring and control of reactor water chemistry consistent with the recommended program elements in Chapter XI.M2, "Water Chemistry," of the most recently issued version of the GALL Report.

3. ***Parameters Monitored/Inspected***: The program manages the following age-related degradation effects and mechanisms that are applicable in general to the RVI components at the facility: (a) cracking induced by stress corrosion cracking (SCC), primary water stress corrosion cracking (PWSCC), irradiation assisted stress corrosion cracking (IASCC), or fatigue/cyclical loading; (b) loss of material induced by wear; (c) loss of fracture toughness

induced by either thermal aging, neutron irradiation embrittlement, and/or void swelling; (d) dimensional changes induced by void swelling and irradiation growth, distortion or deflection; and (e) loss of preload caused by thermal and irradiation-enhanced stress relaxation or creep. For the management of cracking, the program monitors for evidence of surface breaking linear discontinuities if a visual inspection technique is used as the non-destruction examination (NDE) method, or for relevant flaw presentation signals if a volumetric UT method is used as the NDE method. For the management of loss of material, the program monitors for gross or abnormal surfaces conditions that may be indicative of loss of material occurring in the components. For the management of loss of preload, the program monitors for gross surface conditions that may be indicative of loosening in applicable bolted, fastened, keyed, or pinned connections. The program does not directly monitor for loss of fracture toughness that is induced by either thermal aging; neutron irradiation embrittlement, or by void swelling and irradiation growth; instead, the impact of loss of fracture toughness on component integrity is indirectly managed by using visual or volumetric examination techniques to monitor for cracking in the components and by applying applicable reduced fracture toughness properties in the flaw evaluations if cracking is detected in the components and is extensive enough to warrant a supplemental flaw growth or flaw tolerance evaluation under the MRP-227 guidance or ASME Code, Section XI requirements.

Specifically, the program implements the parameters monitored/inspected criteria for [*as an administrative action item for the AMP, applicant is to select one of the following to finish of the sentence, as applicable to its NSSS vendor for its internals: "for B&W designed Primary Components in Table 4-1 of MRP-227"; "for CE designed Primary Components in Table 4-2 of MRP-227"; and for Westinghouse designed Primary Components in Table 4-3 of MRP-227"*]. Additionally, the program implements the parameters monitored/inspected criteria for [*as an administrative action item for the AMP, applicant is to select one of the following to finish of the sentence, as applicable to its NSSS vendor for its internals: "for B&W designed Expansion Components in Table 4-4 of MRP-227"; for CE designed Expansion Components in Table 4-5 of MRP-227"; and for Westinghouse designed Expansion Components in Table 4-6 of MRP-227"*]. The parameters monitored/inspected for Existing Program Components follow the bases for referenced Existing Programs, such as the requirements for ASME Code Class RVI components in ASME Code, Section XI, Table IWB-2500-1, Examination Categories B-N-3, as implemented through the applicant's ASME Code, Section XI program, or the recommended program for inspecting Westinghouse-designed flux thimble tubes in GALL Chapter XI.M37, "Flux Thimble Tube Inspection." No inspections, except for those specified in ASME Code, Section XI, are required for components that are identified as requiring "No Additional Measures," in accordance with the analyses reported in MRP-227.

**4. *Detection of Aging Effects:*** The detection of aging effects is covered in two places: (a) the guidance in Section 4 of MRP-227 provides an introductory discussion and justification of the examination methods selected for detecting the aging effects of interest; and (b) standards for examination methods, procedures, and personnel are provided in a companion document, MRP-228. In all cases, well-established methods were selected. These methods include volumetric UT examination methods for detecting flaws in bolting and various visual (VT-3, VT-1, and EVT-1) examinations for detecting effects ranging from general conditions to detection and sizing of surface-breaking discontinuities.

Cracking caused by SCC, IASCC, and fatigue is monitored/inspected by either VT-1 or EVT-1 examination (for internals other than bolting) or by volumetric UT examination (bolting). The VT-3 visual methods may be applied for the detection of cracking only when the flaw tolerance of the component or affected assembly, as evaluated for reduced fracture toughness properties, is known and has been shown to be tolerant of easily detected large flaws, even under reduced fracture toughness conditions. In addition, VT-3 examinations are used to monitor/inspect for loss of material induced by wear and for general aging conditions, such as gross distortion caused by void swelling and irradiation growth or by gross effects of loss of preload caused by thermal and irradiation-enhanced stress relaxation and creep.

In addition, the program adopts the recommended guidance in MRP-227 for defining the Expansion criteria that need to be applied to inspections of Primary Components and Existing Requirement Components and for expanding the examinations to include additional Expansion Components. As a result, inspections performed on the RVI components are performed consistent with the inspection frequency and sampling bases for Primary Components, Existing Requirement Components, and Expansion Components in MRP-227, which have been demonstrated to be in conformance with the inspection criteria, sampling basis criteria and sample Expansion criteria in Section A.1.2.3.4 of NRC Branch Position RLSB-1. Specifically, the program implements the parameters monitored/inspected criteria and bases for inspecting the relevant parameter conditions for [*as an administrative action item for the AMP, applicant is to select one of the following to finish of the sentence, as applicable to its NSSS vendor for its internals: "B&W designed Primary Components in Table 4-1 of MRP-227"; "CE designed Primary Components in Table 4-2 of MRP-227;" or "Westinghouse designed Primary Components in Table 4-3 of MRP-227"*] and for [*as an administrative action item for the AMP, applicant is to select one of the following to finish of the sentence, as applicable to its NSSS vendor for its internals: "for B&W designed Expansion Components in Table 4-4 of MRP-227;" "for CE designed expansion components in Table 4-5 of MRP-227;" and "for Westinghouse designed Expansion Components in Table 4-6 of MRP-227"*].

The program is supplemented by the following plant-specific Primary Component and Expansion Component inspections for the program (as applicable): [*As a relevant license renewal applicant action item, the applicant is to list (using criteria in MRP-227) each additional RVI component that needs to be inspected as an additional plant-specific Primary Component for the applicant's program and each additional RVI component that needs to be inspected as an additional plant-specific Expansion Component for the applicant's program. For each plant specific component added as an additional primary or Expansion Component, the list should include the applicable aging effects that will be monitored for, the inspection method or methods used for monitoring, and the sample size and frequencies for the examinations*].

In addition, in some cases (as defined in MRP-227), physical measurements are used as supplemental techniques to manage for the gross effects of wear, loss of preload due to stress relaxation, or for changes in dimension due to void swelling, deflection or distortion. The physical measurements methods applied in accordance with this program include [*MRP to input physical measure methods identified by the MRP in response to NRC RAI No. 11 in the NRC's Request for Additional Information to the Mr. Christen B. Larson, EPRI MRP on Topical Report MRP-227 dated November 12, 2009*].

**5. *Monitoring and Trending:*** The methods for monitoring, recording, evaluating, and trending the data that result from the program's inspections are given in Section 6 of MRP-227 and its subsections. The evaluation methods include recommendations for flaw depth sizing and for crack growth determinations as well for performing applicable limit load, linear elastic and elastic-plastic fracture analyses of relevant flaw indications. The examinations and re-examinations required by the MRP-227 guidance, together with the requirements specified in MRP-228 for inspection methodologies, inspection procedures, and inspection personnel, provide timely detection, reporting, and corrective actions with respect to the effects of the age-related degradation mechanisms within the scope of the program. The extent of the examinations, beginning with the sample of susceptible PWR internals component locations identified as Primary Component locations, with the potential for inclusion of Expansion Component locations if the effects are greater than anticipated, plus the continuation of the Existing Programs activities, such as the ASME Code, Section XI, Examination Category B-N-3 examinations for core support structures, provides a high degree of confidence in the total program.

**6. *Acceptance Criteria:*** Section 5 of MRP-227 provides specific examination acceptance criteria for the Primary and Expansion Component examinations. For components addressed by testing programs referenced to ASME Code, Section XI, the IWB-3500 acceptance criteria apply. For other components covered by Existing Programs, the examination acceptance criteria are described within the Existing Program reference document.

The guidance in MRP-227 contains three types of examination acceptance criteria:

• For visual examination (and surface examination as an alternative to visual examination), the examination acceptance criterion is the absence of any of the specific, descriptive relevant conditions; in addition, there are requirements to record and disposition surface breaking indications that are detected and sized for length by VT-1/EVT-1 examinations;

• For volumetric examination, the examination acceptance criterion is the capability for reliable detection of indications in bolting, as demonstrated in the examination Technical Justification; in addition, there are requirements for system-level assessment of bolted or pinned assemblies with unacceptable volumetric (UT) examination indications that exceed specified limits; and

• For physical measurements, the examination acceptance criterion for the acceptable tolerance in the measured differential height from the top of the plenum rib pads to the vessel seating surface in B&W plants are given in Table 5-1 of MRP-227. The acceptance criterion for physical measurements performed on the height limits of the Westinghouse-designed hold-down springs are [*The incorporation of this sentence is a license renewal applicant action item for Westinghouse PWR applicants only – insert the applicable sentence incorporating the specified physical measurement criteria only if the applicant's facility is based on a Westinghouse NSSS design: the Westinghouse applicant is to incorporate the applicable language and then specify the fit up limits on the hold down springs, as established on a plant-specific basis for the design of the hold-down springs at the applicant's Westinghouse-designed facility*].

**7. *Corrective Actions:*** Corrective actions following the detection of unacceptable conditions are fundamentally provided for in each plant's corrective action program. Any detected conditions that do not satisfy the examination acceptance criteria are required to be dispositioned through the plant corrective action program, which may require repair, replacement, or analytical evaluation for continued service until the next inspection. The disposition will ensure that design basis functions of the reactor internals components will continue to be fulfilled for all licensing basis loads and events. Examples of methodologies that can be used to analytically disposition unacceptable conditions are found in the ASME Code, Section XI or in Section 6 of MRP-227. Section 6 of MRP-227 describes the options that are available for disposition of detected conditions that exceed the examination acceptance criteria of Section 5 of the report. These include engineering evaluation methods, as well as supplementary examinations to further characterize the detected condition, or the alternative of component repair and replacement procedures. The latter are subject to the requirements of the ASME Code, Section XI. The implementation of the

### XI.M17 Flow Accelerated Corrosion

Comment/Basis:

XI.M17 – 1 The FAC program is limited to Rev. 2 or 3 rather than "Rev. 2 or later" as recommended by NEI. This will create exceptions to later versions. Credit for NRC Staff review of later versions and acceptance in a safety evaluation report could eliminate exception.

XI.M17 – 2 Element 4 states wall thickness measurements are performed every outage. This may not be true in the future as piping replacements reduce the amount and frequency of inspections.

### 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, ~~or~~ R3 or later revisions (reviewed and accepted by the NRC in a safety evaluation report) 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. NSAC-202L-R2 or R3 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 or R3 to satisfy the criteria specified in 10 CFR Part 50, Appendix B, for development of procedures and control of special processes.

**Evaluation and Technical Basis**

**1. *Scope of Program:*** The FAC program, described by the EPRI guidelines in NSAC-202L-R2, ~~or~~ R3 or later revisions (reviewed and accepted by the NRC in a safety evaluation report), 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 89-08.

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

3. ***Parameters Monitored/Inspected:*** The aging management program monitors the effects of loss of material due to wall thinning on the intended function of piping and components by measuring wall thickness.

4. ***Detection of Aging Effects:*** Degradation of piping and components occurs by wall thinning. The inspection program delineated in NSAC-202L-R2, ~~or~~ R3 or later revisions (reviewed and accepted by the NRC in a safety evaluation report) consists of identification of susceptible locations as indicated by operating conditions or special considerations. Ultrasonic or radiographic testing is used to detect wall thinning. A representative sample of components is selected based on the most susceptible locations for wall thickness measurements at a frequency that meets NSAC 202L requirements ~~every refueling outage~~. The extent and schedule of the inspections ensure detection of wall thinning before the loss of intended function.

5. ***Monitoring and Trending:*** CHECWORKS or a similar predictive code is used to predict component degradation in the systems conducive to FAC, as indicated by specific plant data, including material, hydrodynamic, and operating conditions. CHECWORKS is acceptable because it provides a bounding analysis for FAC. The analysis is bounding because in general the predicted wear rates and component thicknesses are conservative when compared to actual field measurements. CHECWORKS was developed and benchmarked by comparing CHECWORKS predictions against actual measured component thickness measurements obtained from many plants. The inspection schedule developed by the licensee on the basis of the results of such a predictive code provides reasonable assurance that structural integrity will be maintained between inspections. Inspection results are evaluated to determine if additional inspections are needed to ensure that the extent of wall thinning is adequately determined, ensure that intended function will not be lost, and identify corrective actions. Previous wear rate predictions due to FAC may change after a power uprate is implemented. Wear rates are updated in CHECWORKS according to power uprate conditions. Subsequent field measurements are used to calibrate or benchmark the predicted wear rates.

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

7. ***Corrective Actions:*** Prior to service, components for which the acceptance criteria are not satisfied are reevaluated, repaired, or replaced. Long-term corrective actions could include adjusting operating parameters or selecting materials resistant to FAC. As discussed in the appendix to this report, the staff finds the requirements of 10 CFR Part 50, Appendix B, acceptable to address the corrective actions.

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

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

10. ***Operating Experience:*** Wall-thinning problems in single-phase systems have occurred in feedwater and condensate systems (NRC IE Bulletin No. 87-01; NRC Information Notice [IN] 81-28, IN 92-35, IN 95-11, IN 2006-08) and in two-phase piping in extraction steam lines (NRC IN 89-53, IN 97-84) and moisture separation reheater and feedwater heater drains (NRC IN 89-53, IN 91-18, IN 93-21, IN 97-84). Observed wall thinning may be due to mechanisms other than FAC, which require alternate materials to resolve the issue (Licensee Event Report 50-237/2007-003-00). Operating experience shows that the present program, when properly implemented, is effective in managing FAC in high-energy carbon steel piping and components.

**References**

10 CFR Part 50, Appendix B, *Quality Assurance Criteria for Nuclear Power Plants*, Office of the Federal Register, National Archives and Records Administration, 2009.
10 CFR Part 50.55a, *Codes and Standards*, Office of the Federal Register, National Archives and Records Administration, 2009.April 2010 DRAFT XI M17-3 NUREG-1801, Rev. 2.
NRC Generic Letter 89-08, *Erosion/Corrosion-Induced Pipe Wall Thinning*, U.S. Nuclear Regulatory Commission, May 2, 1989.
NRC IE Bulletin 87-01, *Thinning of Pipe Walls in Nuclear Power Plants*, U.S. Nuclear Regulatory Commission, July 9, 1987.
NRC Information Notice 89-53, *Rupture of Extraction Steam Line on High Pressure Turbine*, U.S. Nuclear Regulatory Commission, June 13, 1989.
NRC Information Notice 91-18, *High-Energy Piping Failures Caused by Wall Thinning*, U.S. Nuclear Regulatory Commission, March 12, 1991.
NRC Information Notice 91-18, Supplement 1, *High-Energy Piping Failures Caused by Wall Thinning*, U.S. Nuclear Regulatory Commission, December 18, 1991.
NRC Information Notice 92-35, *Higher than Predicted Erosion/Corrosion in Unisolable Reactor Coolant Pressure Boundary Piping inside Containment at a Boiling Water Reactor*, U.S. Nuclear Regulatory Commission, May 6, 1992.
NRC Information Notice 93-21, *Summary of NRC Staff Observations Compiled during Engineering Audits or Inspections of Licensee Erosion/Corrosion Programs*, U.S. Nuclear Regulatory Commission, March 25, 1993.
NRC Information Notice 95-11, *Failure of Condensate Piping Because of Erosion/Corrosion at a Flow Straightening Device*, U.S. Nuclear Regulatory Commission, February 24, 1995.
NRC Information Notice 97-84, *Rupture in Extraction Steam Piping as a Result of Flow-Accelerated Corrosion*, U.S. Nuclear Regulatory Commission, December 11, 1997.

NSAC-202L-R2, *Recommendations for an Effective Flow Accelerated Corrosion Program*, Electric Power Research Institute, Nuclear Safety Analysis Center, Palo Alto, CA, April 8, 1999.
NSAC-202L-R3, *Recommendations for an Effective Flow Accelerated Corrosion Program*, (1011838), Electric Power Research Institute, Nuclear Safety Analysis Center, Palo Alto, CA, May 2006.
NUREG-1344, *Erosion/Corrosion-Induced Pipe Wall Thinning in U.S. Nuclear Power Plants*, P. C. Wu, U.S. Nuclear Regulatory Commission, April 1989.
NRC Information Notice 2006-08, *Secondary Piping Rupture at the Mihama Power Station in Japan*, U.S. Nuclear Regulatory Commission, March 16, 2006.

NRC Licensee Event Report 50-237/2007-003-00, *Unit 2 High Pressure Coolant Injection System Declared Inoperable*, U.S. Nuclear Regulatory Commission, September 24, 2007.

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## XI.M18 BOLTING INTEGRITY

Comment/Basis

XI.M18-1 Revise first paragraph of Program Description for clarity.

XI.M18-2 Currently the Program Description says this program doesn't apply to structural bolts and the SCOPE says this program doesn't apply to the Reactor Vessel closure studs. Both exceptions should be listed both places.

XI.M18-3 The third sentence in the first paragraph is redundant to the next paragraph. Suggest this sentence be deleted.

XI.M18 -4 In the third paragraph of the program description need to include , XI.S6, Structures Monitoring and XI. S7, RG 1.127, Inspection of Water-Control Structures Associated with Nuclear Power Plants which both manage structural bolting.

XI.M18 – 5 In elements 3 and 4, high strength bolts are described. In element 2, preventive actions, state that use of high strength bolts is avoided. Reference to high strength bolting should be removed as it is only an issue for structural bolting which has been removed..

XI.M18 – 6 Elements 1, 3, 4, 6 and 7. The main change is that all the structural bolting has been pulled out of the Bolting Integrity program and inserted in IWE, IWF, Structures Monitoring, and RG 1.127. Although removing structural bolting from the Bolting Integrity program would simplify Bolting Integrity (a little bit), it would unnecessarily complicate these others AMPS.

## XI.M18 BOLTING INTEGRITY

**Program Description**

The program manages aging of ~~focuses on~~ closure bolting for pressure retaining components. The program ~~and~~ relies on recommendations for a comprehensive bolting integrity program, as delineated in the following documents:

NUREG-1339, "Resolution of Generic Safety Issue 29: Bolting Degradation or Failure in Nuclear Power Plants"

~~and industry recommendations, as delineated in the~~ Electric Power Research Institute (EPRI) NP-5769, "Degradation and Failure of Bolting in Nuclear Power Plants" (with the exceptions noted in NUREG-1339 for safety-related bolting), and

~~. The program also relies on industry recommendations for comprehensive bolting maintenance, as delineated in~~ EPRI TR-104213, "Bolted Joint Maintenance & Application Guide."

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.

Aging management program (AMP) XI.M1, "ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD," includes inspection of safety-related and non-safety-related closure bolting and supplements this bolting integrity program. AMP XI.M3, "Reactor Head Closure Stud Bolting" manages aging of the reactor head closure bolting. AMPs XI.S1, "ASME Section XI, Subsection IWE"; XI.S3, "ASME Section XI, Subsection IWF"; XI.S6, "Structures Monitoring," XI. S7, "RG 1.127, Inspection of Water-Control Structures Associated with Nuclear Power Plants" and XI.M23, "Inspection of Overhead Heavy Load and Light Load (Related to Refueling) Handling Systems," manage ~~inspection~~ aging of safety-related and non-safety related structural bolting.

**Evaluation and Technical Basis**

1. ***Scope of Program:*** This program ~~covers~~ manages aging of closure bolting for pressure retaining components within the scope of license renewal, including both safety-related and non-safety-related bolting. The~~is~~ program does not manage aging ~~management~~ of reactor head closure stud bolting (~~is addressed by~~ AMP XI.M3) or structural bolting (AMPs XI.S1 and XI.S3)~~, "Reactor Head Closure Stud Bolting," and is not included in this program~~.

2. ***Preventive Actions:*** Selection of bolting material and the use of lubricants and sealants is in accordance with the guidelines of EPRI NP-5769 and the additional recommendations of NUREG-1339 to prevent or mitigate degradation and failure of safety-related bolting. NUREG-1339 takes exception to certain items in EPRI NP-5769 and recommends additional measures with regard to them. Of particular note, use of molybdenum disulfide ($MoS_2$) as a lubricant has been shown to be a potential contributor to stress corrosion cracking (SCC) and should not be used. ~~Preventive measures also include using bolting material that has an actual measured yield strength limited to less than 1,034 megapascals (MPa) (150 kilo-pounds per square inch [ksi])~~. Bolting replacement activities include proper torquing of the bolts and checking for uniformity of the gasket compression after assembly. Maintenance practices require the application of an appropriate preload based on guidance in EPRI documents, manufacturer recommendations, or engineering evaluation.

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 or prestress. Bolting for other pressure retaining components is inspected for signs of leakage. ~~High strength closure bolting (actual yield strength greater than or equal to 1,034 MPa [150 ksi]), if used, should be monitored for cracking.~~

4. ***Detection of Aging Effects:*** The American Society of Mechanical Engineers (ASME) Section XI Inservice Inspection, Subsections IWB, IWC, and IWD program implements inspection of Class 1, Class 2, and Class 3 pressure retaining bolting in accordance with requirements of ASME Code Section XI, Tables IWB-2500-1, IWC-2500-1, and IWD-2500-1. These include volumetric and visual (VT-1) examinations, as appropriate. In addition, for both ASME Code class bolting and non-ASME Code class bolting, periodic system walkdowns and inspections (at least once per refueling cycle) ensure detection of leakage at bolted joints before the leakage becomes excessive. Bolting inspections should include consideration of the guidance applicable for pressure boundary bolting in NUREG-1339, ~~and in~~ EPRI NP-5769 and EPRI TR-104213.

   Degradation of pressure boundary closure bolting due to crack initiation, loss of preload, or loss of material may result in leakage from the mating surfaces or joint connections of pressure boundary components. Periodic inspection of pressure boundary components for signs of leakage ensures that age-related degradation of closure bolting is detected and corrected before component leakage becomes excessive. Accordingly, pressure retaining bolted connections should be inspected at least once per refueling cycle. The inspections may be performed as part of ASME Code Section XI leakage tests or as part of other periodic inspection activities such as system walkdowns or an external surfaces monitoring program.

   ~~High strength closure bolting (actual yield strength greater than or equal to 1,034 MPa (150 ksi), may be subject to stress corrosion cracking. For high strength closure bolts (regardless of code classification), volumetric examination in accordance to that of ASME Code Section XI, Table IWB-2500-1, Examination Category B-G-1 should be performed.~~

5. ***Monitoring and Trending:*** The inspection schedules of ASME Section XI components are effective and ensure timely detection of applicable aging effects. If a bolting connection for pressure retaining components not covered by ASME Section XI is reported to be leaking, it may be inspected daily or in accordance with the corrective action process. If the leak rate is increasing, more frequent inspections may be warranted.

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, 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-ASME code class bolting) is performed in accordance with the guidelines and recommendations of EPRI TR-104213. 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:*** Site quality assurance 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 confirmation process and administrative controls.

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 threaded bolting and fasteners in closures for the reactor coolant pressure boundary has occurred from boric acid corrosion, SCC, and fatigue loading (U.S. Nuclear Regulatory Commission [NRC] IE Bulletin 82-02, NRC Generic Letter 91-17). SCC has occurred in high strength bolts used for nuclear steam supply system component supports (EPRI NP-5769). The bolting integrity program developed and implemented in accordance with the applicant's docketed responses to NRC communications on bolting events have provided an effective means of ensuring bolting reliability. These programs are documented in EPRI NP-5769 and TR-104213 and represent industry consensus.

    Degradation related failures have occurred in downcomer Tee-quencher bolting in boiling water reactors (BWRs) designed with drywells (ADAMS Accession Number ML050730347). Leakage from bolted connections has been observed in reactor building closed cooling systems of BWRs (LER 50-341/2005-001).

    The applicant is to evaluate applicable operating experience to support the conclusion that the effects of aging are adequately managed.

**References**

10 CFR Part 50, Appendix B, *Quality Assurance Criteria for Nuclear Power Plants*, Office of the Federal Register, National Archives and Records Administration, 2009.

10 CFR 50.55a, *Codes and Standards*, Office of the Federal Register, National Archives and Records Administration, 2009.

ASME Section XI, *Rules for Inservice Inspection of Nuclear Power Plant Components*, The ASME Boiler and Pressure Vessel Code, 2004 edition as approved in 10 CFR 50.55a11[5], The American Society of Mechanical Engineers, New York, NY.

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

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

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

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

NRC Morning Report, *Failure of Safety/Relief Valve Tee-Quencher Support Bolts*, March 14, 2005. (ADAMS Accession Number ML050730347)

[5] Refer to the GALL Report, Chapter I, for applicability of other editions of the ASME Code, Section XI.

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

## XI.M19 STEAM GENERATORS

Comment/Basis:

XI.M19-1 Maintaining water chemistry should be left to the Water Chemistry Program, XI.M2, as it was in other programs. Revise PREVENTIVE ACTIONS accordingly.

XI.M19-2 Program Element 2, Preventive Actions, lists chemical cleaning as a secondary side maintenance activity. Recommend to remove chemical cleaning since it is not done routinely since it is very expensive and can cause harm to tube materials if not done properly. Recommend to leave in sludge lancing as a secondary side maintenance activity. Recommend to state that secondary chemistry programs may be enhanced to add chemicals or adjust chemistry as needed to minimize deposition onto tubes (i.e., adding a dispersant such as polyacrylic acid [PAA]) as a preventive action. See Below

## XI.M19 STEAM GENERATORS

### PROGRAM DESCRIPTION

The Steam Generator program is applicable to managing the aging of steam generator tubes, plugs, sleeves, and secondary side components that are contained within the steam generator (i.e., secondary side internals).

The establishment of a steam generator program for ensuring steam generator tube integrity is required by plant technical specifications. The steam generator tube integrity portion of the technical specifications at each PWR contains the same fundamental requirements as outlined in the standard technical specifications of NUREG-1430, Volume 1, Rev. 3, for Babcock & Wilcox pressurized water reactors (PWRs); NUREG-1431, Volume 1, Rev. 3, for Westinghouse PWRs; and NUREG-1432, Volume 1, Rev. 3, for Combustion Engineering PWRs. The requirements pertaining to steam generators in these three versions of the standard technical specifications are essentially identical. The technical specifications require tube integrity to be maintained and specify performance criteria, condition monitoring requirements, inspection scope and frequency, acceptance criteria for the plugging or repair of flawed tubes, acceptable repair methods, and leakage monitoring requirements.

The nondestructive examination techniques used to inspect tubes, plugs, sleeves, and secondary side internals are intended to identify components (e.g., tubes, plugs) with degradation that may need to be removed from service or repaired.

The Steam Generator program at PWRs is modeled after Nuclear Energy Institute (NEI) 97-06, Revision 2, "Steam Generator Program Guidelines." This program references a number of industry guidelines (e.g., the EPRI PWR Steam Generator Examination Guidelines, PWR Primary-to-Secondary Leak Guidelines, PWR Primary Water Chemistry Guidelines, PWR Secondary Water Chemistry Guidelines, Steam Generator Integrity Assessment Guidelines, Steam Generator In Situ Pressure Test Guidelines) and incorporates a balance of prevention, mitigation, 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 steam generator plugs, sleeves, and secondary side internals. Therefore, all these components are addressed by this aging management program (AMP). The NEI 97-06 program has been effective at managing the aging effects associated with steam generator tubes, plugs, sleeves, and secondary side internals.

**Evaluation and Technical Basis**

1. ***Scope of Program:*** This program addresses degradation associated with steam generator tubes, plugs, sleeves, and secondary side components that are contained within the steam generator (i.e., secondary side internals). It does not cover degradation associated with the steam generator shell, channelhead, nozzles, or the welds associated with these components.

2. ***Preventive Actions:*** This program includes preventive and mitigative actions for addressing degradation. Preventive and mitigative measures that are part of the Steam Generator program include ~~primary- and secondary-side water chemistry programs~~ foreign material exclusion programs, and other primary and secondary side maintenance activities. ~~The water chemistry program for PWRs relies on monitoring and control of reactor water chemistry based on industry guidelines such as EPRI 1014986 (PWR Primary Water Chemistry Guidelines-Revision 6) and EPRI 1016555 (PWR Secondary Water Chemistry Guidelines-Revision 7). 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.~~ The program ~~also~~ includes foreign material exclusion as a means to inhibit wear degradation and secondary side maintenance activities such as sludge lancing and chemical cleaning for removing deposits that may contribute to degradation. Guidance on foreign material exclusion is provided in NEI 97-06. Guidance on maintenance of secondary side integrity is provided in the EPRI Steam Generator Integrity Assessment Guidelines. Primary side preventive maintenance activities include replacing plugs made with corrosion susceptible materials with more corrosion resistant materials and preventively plugging tubes susceptible to degradation.

   Extensive deposit buildup in the steam generators ~~can be indicative of a condition that~~ could affect tube integrity. The EPRI Steam Generator Integrity Assessment Guidelines, which are referenced in NEI 97-06, provide guidance on maintenance on the secondary side of the steam generator, including secondary side cleaning.

   Maintaining high water purity reduces susceptibility to SCC or IGSCC. . Reactor coolant water chemistry is monitored and maintained in accordance with the Water Chemistry Program. The program description and evaluation and technical basis of monitoring and maintaining reactor water chemistry are addressed in Section XI.M2, "Water Chemistry."

3. ***Parameters Monitored/Inspected:*** There are currently three types of steam generator tubing used in the United States: mill annealed Alloy 600, thermally treated Alloy 600, and thermally treated Alloy 690. Mill annealed Alloy 600 steam generator tubes have experienced degradation due to corrosion (e.g., primary water stress corrosion cracking, outside diameter stress corrosion cracking, intergranular attack, pitting, and wastage) and mechanically induced phenomena (e.g., denting, wear, impingement damage, and fatigue). Thermally treated Alloy 600 steam generator tubes have experienced degradation due to

corrosion (primarily cracking) and mechanically induced phenomena (primarily wear). Thermally treated Alloy 690 tubes have only experienced tube degradation due to mechanically induced phenomena (primarily wear). Degradation of tube plugs, sleeves, and secondary side internals have also been observed depending, in part, on the material of construction of the specific component.

The program includes an assessment of the forms of degradation to which a component is susceptible and implementation of inspection techniques capable of detecting those forms of degradation. The parameter monitored is specific to the component and the acceptance criteria for the inspection. For example, the severity of tube degradation may be evaluated in terms of the depth of degradation or measured voltage, dependent on whether a depth-based or voltage-based tube repair criteria (acceptance criteria) is being implemented for that specific degradation mechanism. Other parameters monitored include signals of excessive deposit buildup (e.g., steam generator water level oscillations), which may result in fatigue failure of tubes or corrosion of the tubes; water chemistry parameters, which may indicate unacceptable levels of impurities; primary-to-secondary leakage, which may indicate excessive tube, plug, or sleeve degradation; and the presence of loose parts or foreign objects on the primary and secondary side of the steam generator, which may result in tube damage.

Water chemistry parameters are also monitored as discussed in Chapter XI.M2. The EPRI PWR Steam Generator Primary-to-Secondary Leakage Guidelines (EPRI 1008219) provides guidance on monitoring primary-to-secondary leakage. The EPRI Steam Generator Integrity Assessment Guidelines (EPRI 1012987) provide guidance on secondary side activities.

In summary, the NEI 97-06 program provides guidance on parameters to be monitored or inspected.

4. ***Detection of Aging Effects:*** The technical specifications require a Steam Generator program be established and implemented to ensure steam generator tube integrity is maintained. This requirement ensures that components that could compromise tube integrity are properly evaluated or monitored (e.g., degradation of a secondary side component that could result in a loss of tube integrity is managed by this program). The inspection requirements in the technical specifications are intended to detect tube and sleeve degradation (i.e., aging effects), if they should occur.

   The technical specifications are performance-based, and the actual scope of the inspection and the expansion of sample inspections are justified based on the results of the inspections. The goal is to perform inspections at a frequency sufficient to provide reasonable assurance of steam generator tube integrity for the period of time between inspections.

   The general condition of some components (e.g., plugs and secondary side components) may be monitored visually and subsequently, more detailed inspections may be performed if degradation is detected.

   NEI 97-06 provides additional guidance on inspection programs to detect degradation of tubes, sleeves, plugs, and secondary side internals. The frequencies of the inspections are based on technical assessments. Guidance on performing these technical assessments is contained in NEI 97-06 and the associated industry guidelines.

   The inspections and monitoring are performed by qualified personnel using qualified techniques in accordance with approved licensee procedures. The EPRI PWR Steam

Generator Examination Guidelines (EPRI 1013706) contains guidance on the qualification of steam generator tube inspection techniques.

The primary-to-secondary leakage monitoring program provides a potential indicator of a loss of steam generator tube integrity. NEI 97-06 and the associated EPRI guidelines provide information pertaining to an effective leakage monitoring program.

5. ***Monitoring and Trending:*** Condition monitoring assessments are performed to determine whether the structural and accident induced leakage performance criteria were satisfied during the prior operating interval. Operational assessments are performed to verify that structural and leakage integrity will be maintained for the planned operating interval before the next inspection. If tube integrity can not be maintained for the planned operating interval before the next inspection, corrective actions are taken in accordance with the plant's corrective action program. Comparisons of the results of the condition monitoring assessment to the predictions of the previous operational assessment are performed to evaluate the adequacy of the previous operational assessment methodology. If the operational assessment was not conservative in terms of the number and/or severity of the condition, corrective actions are taken in accordance with the plant's corrective action program.

   For tubes and sleeves, the technical specifications require condition monitoring and operational assessments to be performed (although the technical specifications do not explicitly require operational assessments, they are required implicitly by the need to maintain tube integrity for the period of time between inspections). Condition monitoring and operational assessments are done in accordance with the technical specification requirements and guidance in NEI 97-06 and the EPRI Steam Generator Integrity Assessment Guidelines.

   The goal of the inspection program for all components covered by this AMP is to ensure the components continue to function consistent with the design and licensing basis of the facility (including regulatory safety margins).

   Assessments of the degradation of steam generator secondary side internals are performed in accordance with the guidance in the EPRI Steam Generator Integrity Assessment Guidelines to ensure technical specification requirements are satisfied.

6. ***Acceptance Criteria:*** Assessment of tube and sleeve integrity and plugging or repair criteria of flawed and sleeved tubes is in accordance with plant technical specifications. The criteria for plugging or repairing steam generator tubes and sleeves are based on U.S. Nuclear Regulatory Commission (NRC) Regulatory Guide 1.121 and are incorporated into plant technical specifications. Guidance on assessing the acceptability of flaws also is provided in NEI 97-06 and the associated EPRI guidelines, including the EPRI Steam Generator In-Situ Pressure Test Guidelines and EPRI Steam Generator Integrity Assessment Guidelines.

   Degraded plugs, degraded secondary side internals, and leaving a loose part or a foreign objects in the steam generator are evaluated for continued acceptability on a case-by-case basis. NEI 97-06 and the associated EPRI guidelines provide guidance on the performance of these evaluations. The intent of these evaluations is to ensure the components affected by parts or objects have adequate integrity consistent with the design and licensing basis of the facility.

Guidance on the acceptability of primary-to-secondary leakage and water chemistry parameters also are discussed in NEI 97-06 and the associated EPRI guidelines.

7. ***Corrective Actions:*** For degradation of steam generator tubes and sleeves (if applicable), the technical specifications provide requirements on the actions to be taken when the acceptance criteria are not met. For degradation of other components, the appropriate corrective action is evaluated per NEI 97-06 and the associated EPRI guidelines, the American Society of Mechanical Engineers (ASME) Code Section XI (2004 Edition)[6], 10 CFR 50.65, and 10 CFR Part 50, Appendix B, as appropriate. As discussed in the appendix to this report, the staff finds the requirements of 10 CFR Part 50, Appendix B, acceptable for ensuring effective corrective actions.

8. ***Confirmation Process:*** Site quality assurance (QA) procedures, review and approval processes, and Site quality assurance 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 confirmation process and administrative controls.

   In addition, the adequacy of the preventive measures in the Steam Generator program is confirmed through periodic inspections.

9. 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:*** Several generic communications have been issued by the NRC related to the steam generator programs implemented at plants. The reference section lists many of these generic communications. In addition, NEI 97-06 provides guidance to the industry for routinely sharing pertinent steam generator operating experience and for incorporating lessons learned from plant operation into guidelines referenced in NEI 97-06. The latter includes providing interim guidance to the industry, when needed.

    The NEI 97-06 program has been effective at managing the aging effects associated with steam generator tubes, plugs, sleeves, and secondary side components that are contained within the steam generator (i.e., secondary side internals) such that the steam generators can perform their intended safety function.

**References**

10 CFR Part 50, Appendix B, *Quality Assurance Criteria for Nuclear Power Plants*, Office of the Federal Register, National Archives and Records Administration, 2009.

10 CFR Part 50.55a, *Codes and Standards*, Office of the Federal Register, National Archives and Records Administration, 2009.

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, 2009.

EPRI 1008219, *PWR Primary-to-Secondary Leak Guidelines: Revision 3*, Electric Power Research Institute, Palo Alto, CA, December 2004.

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[6] Refer to the GALL Report, Chapter 1, for applicability of other editions of the ASME Code.

EPRI 1012987, *Steam Generator Integrity Assessment Guidelines: Revision 2*, Electric Power Research Institute, Palo Alto, CA, July 2006.

EPRI 1013706, *PWR Steam Generator Examination Guidelines: Revision 7*, Electric Power Research Institute, Palo Alto, CA, October 2007.

EPRI 1014983, *Steam Generator In-Situ Pressure Test Guidelines: Revision 3*, Electric Power Research Institute, Palo Alto, CA, August 2007.

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**XI.M22 Boraflex Monitoring**

Comment/Basis:

XI.M22 – 1 Five years is not a frequency. Inserted "once every" before "5 years".

**4. *Detection of Aging Effects:*** Aging effects on Boraflex panels are detected by monitoring silica levels in the spent fuel storage pool on a regular basis, such as monthly, quarterly, or annually (depending on Boraflex panel condition); by performing blackness testing to measure gap formation or measuring boron areal density on a frequency determined by the material condition of the Boraflex panels, with a maximum of <u>once every</u> 5 years; and by applying predictive methods to the measured results. The amount of boron carbide present in the Boraflex panel is determined through direct measurement of boron areal density by blackness testing or by periodic verification of boron loss through areal density measurement techniques such as the BADGER device. Frequent Boraflex testing is sufficient to ensure that Boraflex panel degradation does not compromise criticality analysis for the spent fuel pool storage racks. Additionally, changes in the level of silica present in the spent fuel pool water provide an indication of changes in the rate of degradation of Boraflex panels.

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## XI.M24 Compressed Air Monitoring

Comment/Basis:

XI.M24 – 1 Program Description and Element 3, 4 and 5 -Specifying leak testing and compressor cycle time as an aspect of aging management is inappropriate since the presence of leakage confirms that aging effects have not been appropriately managed. In addition leakage testing will normally detect the failure of isolation valve leak by and seal failures that are not passive components. Confirming the presence of moisture and contaminants along with visual inspections will confirm the effectiveness of aging management.

XI.M24 – 2 In element 3 erosion is not an aging effect in GALL for air systems such that there is no need for inspection.

XI.M24 – 3 In element 3 the use of the word "all" is unclear and implies that every component be examined. Recommend rewording to delete "all" and include "when available" to ensure components are inspected when opened for access.

**Program Description**

The purpose of the compressed air monitoring program is to ensure the integrity of the compressed air system. The program consists of monitoring moisture content, corrosion, and performance of the compressed air system. This includes ~~(a) frequent leak testing of the system pressure boundary;~~ (a~~b~~) preventive monitoring of water (moisture) and other potential contaminants to keep within the specified limits; and (b~~c~~) inspection of components for indications of loss of material due to corrosion.

**3.** ***Parameters Monitored/Inspected:*** Maintaining moisture and other corrosive contaminants below acceptable limits is a preventives measure and mitigates loss of material due to corrosion Periodic air samples are taken and analyzed for moisture and other corrosives. Inspections of ~~all~~ accessible internal surfaces when available are performed for signs of corrosion, ~~erosion,~~ and abnormal corrosion products that might indicate a loss of material within the system. ~~Pressure decay leak testing is performed periodically to ensure the integrity of the pressure boundary. Performance monitoring, such as compressor cycle time, provides an indication of system integrity.~~

**4.** ***Detection of Aging Effects:*** Moisture and other corrosives increase the potential for loss of material due to corrosion. The program periodically samples and tests the air quality in the compressed system for moisture in accordance with industry standards, such as ASI –S7.0.01. Typically, compressed systems have in-line dew point instrumentation that is checked daily to ensure moisture content is within specifications. Additionally, periodic visual inspections of critical component internal surfaces (compressors, dryers, after-coolers, and filters) are performed for signs of loss of material due to corrosion. ASME O/M-S/G, Part 17 (1998) provides guidance for inspection frequency and inspection methods of these components. ~~Leaks found during pressure testing are evaluated for potential wall thinning because of the loss of material.~~ If corrosives other than moisture are present, appropriate corrective actions should be taken.

**5. *Monitoring and Trending:*** Daily readings of system dew point are recorded and trended. Air quality analysis results are reviewed to determine if alert levels or limits have been reached or exceeded. This review also checks for unusual trends. ASME O/M-S/G, Part 17, provides guidance for monitoring and trending data. Visual inspection results are compared to previous results to ascertain if adverse long term trends exist. The effects of corrosion are monitored by visual inspection ~~and periodic system and component tests, including leak rate tests on the system and on individual items or components.~~ ~~These tests verify proper operation by comparing measured values of performance with specified performance limits.~~ Test data are analyzed and compared to data from previous tests to provide for the timely detection of aging effects on passive components.

**6. *Acceptance Criteria:*** Acceptance criteria for air quality moisture limits are established based on accepted industry standards, such as ISA-S7.0.01. *Internal surfaces should not show signs* of corrosion (general, pitting, and crevice) that could indicate the potential loss of function of the component. Manufacturers' certifications can be used to demonstrate that the bottled air meets acceptable quality standards. ~~The pressure decay leak tests verify proper operation by comparing measured values of performance with specific performance limits.~~

**XI.M26 Fire Protection**

Comment/Basis:

XI.M26 – 1 Element 1 - states that the program manages the effects of loss of material and cracking, however Increased hardness, shrinkage, and loss of strength are also addressed for elastomer penetration seals in line item VII.G.A-91. Need to add these aging effects

XI.M26 – 2 Elements 3 and 4 - Since inspections of penetration seals are performed in accordance with NRC-approved fire protection program as added by the revision, it is not necessary to specify the quantity of each seal type since this value will be defined by the site specific NRC approved program. This would be consistent with the other changes in the program in element 4 that added that the frequency of the inspection will be in accordance with an NRC approved fire protection program. The quantity of seal type to be inspected is specified in this NRC approved program and does not need to be defined.

XI.M26 – 3 Element 4 - Though the option is available to use a different frequency, consider removing 6 month frequency since it was removed from Element 3 as requested earlier. Inspections of the system are in accordance with NRC-approved fire protection program.

XI.M26 – 4 Element 6 – Typo, remove second "of degradation".

**1. *Scope of Program:*** This program manages the effects of loss of material and cracking, increased hardness, shrinkage, and loss of strength on the intended function of the penetration seals; fire barrier walls, ceilings, and floors; other materials (e.g., flamastic, 3M fire wrapping, spray-on fire proofing material, intumescent coating, etc.) that serve a fire barrier function; and all fire-rated doors (automatic or manual) that perform a fire barrier function. It also manages the aging effects on the intended function of the halon/CO2 fire suppression system.

**3. *Parameters Monitored/Inspected:*** Visual inspection of ~~not less than 10 percent of~~ each type of penetration seal is performed during walkdowns. 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 that are directly caused by increased hardness, and shrinkage of seal material due to loss of material. Visual inspection of the fire barrier walls, ceilings, and floors and other fire barrier materials detects any sign of degradation such as cracking, spalling, and loss of material caused by freeze-thaw, chemical attack, and reaction with aggregates that could affect their intended fire protection function. Fire-rated doors are visually inspected to detect any degradation of door surfaces.

The periodic visual inspection and function test is performed to examine for signs of corrosion that may lead to the loss of material of the halon/CO2 fire suppression system.

**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 ~~not less than 10 percent of~~ each type of seal in walkdowns is performed at a frequency in accordance with an NRC-approved fire protection program (e.g., Technical Requirements Manual, Appendix R program, etc.). 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 and other fire barrier materials are performed in walkdowns at a frequency in accordance with an NRC-approved fire protection program 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.

Visual inspections of the halon/CO2 fire suppression system are performed to detect any sign of corrosion. The periodic functional tests ~~is~~ are performed ~~at least once every 6 months or~~ on a schedule in accordance with an NRC-approved fire protection program. Inspections are performed to detect degradation of the halon/CO2 fire suppression system before the loss of the component intended function.

**6. *Acceptance Criteria:*** Inspection results are acceptable if there are no signs of degradation that could result in the loss of the fire protection capability ~~of degradation~~ due to loss of material. The acceptance criteria include (a) 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; (b) no significant indications of concrete cracking, spalling, and loss of material of fire barrier walls, ceilings, and floors; and in other fire barrier materials; (c) no visual indications of missing parts, holes, and wear; and (d) no deficiencies in the functional tests of fire doors. Also, no indication of excessive loss of material due to corrosion in the halon/CO2 fire suppression system is acceptable.

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**XI. M30 Fuel Oil Chemistry**

Comment/Basis -

XI. M30 – 1 Element 1 and Element 3 - state that the program is focused on managing the loss of material due to general, pitting, and MIC. Issue: The program also manages the aging mechanisms of crevice corrosion and fouling that leads to corrosion. See items VII.H1.AP.105 and VII.H2.AP-105 as examples.

XI.M30 – 2 Element 4 - only allows for periodic multilevel tank sampling. Issue: There are numerous fuel oil tanks that do not have the capability to be sampled using multilevel sampling techniques due to their design (e.g., no top access). Recommend allowing alternate sampling techniques that provide an equivalent conservative sample. For example, a single point tank drain at the lowest point on the tank sample would be considered a more conservative sample. This program samples for water, sediment, and particulate contamination. Water, sediment, and particulate tend to settle towards the bottom of the tank making a true bottom sample more conservative. Previously for Oyster Creek and TMI bottom samples were considered an exception to the program which were accepted by the NRC. This change would eliminate the need to make this an exception.

XI.M30 – 3 Element 4 - identifies the requirements for tank inspections prior to the period of extended operation. It requires each tank to be drained and cleaned, and the internal surfaces visually inspected (if physically possible) and volumetrically inspected. Issue: The requirement for volumetric inspection should only apply if degradation is found during visual inspection. If a visual internal inspection cannot be performed, then, a volumetric examination (from the outside looking in) must be performed in lieu of that visual.

**Evaluation and Technical Basis**

**1. Scope of Program:** Components within the scope of the program are the diesel fuel oil storage tanks, piping, and other metal components subject to aging management review that are exposed to an environment of diesel fuel oil. The program is focused on managing loss of material due to general, pitting, crevice, and microbiologically-influenced corrosion (MIC), and fouling of the diesel fuel tank internal surfaces.

**3. Parameters Monitored/Inspected:** The program is focused on managing loss of material due to general, pitting, crevice, MIC, and fouling of the diesel fuel tank internal surfaces. The aging management program monitors fuel oil quality through receipt testing and periodic sampling of stored fuel oil. Parameters monitored include water and sediment content, total particulate concentration, and the levels of microbiological organisms in the fuel oil. Water and microbiological organisms in the fuel oil storage tank increase the potential for corrosion. Sediment and total particulate content may be indicative of water intrusion or corrosion.

**4. Detection of Aging Effects:** Loss of material due to corrosion of the diesel fuel oil tank or other components exposed to diesel fuel oil cannot occur without exposure of the tank internal surfaces to contaminants in the fuel oil, such as water and microbiological organisms. Periodic multilevel sampling provides assurance that fuel oil contaminants are below unacceptable levels. <u>In situations where tank design features do not allow for multilevel sampling, an equivalent or more conservative sampling approach may be used.</u>

At least once during the 10-year period prior to the period of extended operation, each diesel fuel tank is drained and cleaned, the internal surfaces are visually inspected (if physically possible) and volumetrically-inspected <u>if evidence of degradation is observed during visual inspection, or, if visual inspection is not possible.</u> During the period of extended operation, at least once every 10-years, each diesel fuel tank is drained and cleaned, the internal surfaces are visually inspected (if physically possible), and if evidence of degradation is observed during inspections, <u>or if visual inspection is not possible,</u> these diesel fuel tanks are volumetrically inspected.

Prior to the period of extended operation, a one-time inspection (i.e., Chapter XI.M32) of selected components exposed to diesel fuel oil is performed to verify the effectiveness of the Fuel Oil Chemistry program.

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## XI.M31 REACTOR VESSEL SURVEILLANCE

<u>Comment/Basis:</u>

XI.M31 – 1 In element 5b consistent with RG 1.99 when two or credible surveillance capsules are available then embrittlement may be projected using position 2. Suggest rewording

**5. *Monitoring and Trending:*** The program provides reactor vessel embrittlement data for the time limited aging analyses (TLAAs) on neutron irradiation embrittlement (e.g., upper-shelf energy, pressurized thermal shock and pressure-temperature limits evaluations, etc.) for 60 years. The program is designed to periodically remove and test capsules for monitoring and trending purposes. Refer to the Standard Review Plan for License Renewal, Section 4.2, for the

NRC acceptance criteria and review procedures for reviewing TLAAs for neutron irradiation embrittlement. The TLAAs are projected in accordance with NRC Regulatory Guide (RG) 1.99, Rev. 2, "Radiation Embrittlement of Reactor Vessel Materials," and the pressurized thermal shock rules (10 CFR 50.61 and 10 CFR 50.61a). When using NRC RG 1.99, Rev. 2, or equivalent provisions in 10 CFR 50.61, a licensee has a choice of the following:

(a) Neutron Embrittlement Using Chemistry Tables and Upper Shelf Energy Figures

An applicant may use the tables and figures 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 NRC RG 1.99, Rev. 2.

(b) Neutron Embrittlement Using Surveillance Data
When two or more credible surveillance data are available, the extent of reactor vessel neutron embrittlement for the period of extended operation may ~~must~~ 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 and extended operating term. A plant-specific program or an integrated surveillance program during the period of extended operation provides for the collection of additional data.

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**XI.M32 One – Time Inspection**

Comment/Basis

XI.M32 – 1 Element 4 – This section still does not define the terms qualified and equivalent such that the inspections specified in this program would have to be performed by ASME qualified inspectors using ASME procedures even on non code components. Definitions similar to those proposed in XI.M36 External Surfaces and M38 could be applied in this program such as qualified in accordance with site procedures and programs for the type of examination being performed. See proposed changes below.

**4. Detection of Aging Effects**

The program relies on established NDE techniques, including visual, ultrasonic, and surface techniques. Inspections are performed by qualified personnel qualified in accordance with site procedures and programs to perform the type of examination specified. Examinations should follow~~ing~~ procedures consistent with the American Society of Mechanical Engineers (ASME) Code and 10 CFR Part 50, Appendix B[15] for Code components. For non code components the techniques should follow procedures that include requirements for items such as lighting, presence of protective coatings, and cleaning processes that ensure an adequate exam. In addition, a description of Enhanced Visual Examination (EVT-1) is found in Boiling Water Reactor Vessel and Internals Project (BWRVIP)-03 and Materials Reliability Program (MRP)-228.

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**XI.M35 One-Time Inspection – Small Bore Piping**

Comment/Basis:

XI.M35 – 1 "Vibratory loading" is used to describe an aging mechanism managed by this program in elements 1, 4 and 5. In Chapter IX.F, the definition of fatigue cites "Vibration is generally induced by external equipment operation." By definition, aging induced by external equipment operation is not an aging mechanism but a design issue that will be identified early in plant life and corrected as discussed in recent OE discussions with NEI and the NRC staff and as presented in this program. There is no basis for the claim that this is an aging mechanism since it is not related to the age of the equipment but the impact of external operation. Comment also applies to aging management review tables citing vibratory loading which should also be eliminated. Recommend change to vibration fatigue if this must be retained as an aging mechanism.

XI.M35 – 2 In element 1 and description there are statements regarding program applicability to plants that have performed design changes to mitigate cracking from vibration. Was the intent of this to have a periodic plant specific program if the design changes have not eliminated the vibration issue such that cracking has reoccurred, and if design changes have been implemented without additional cracking that XI.M35 is to be used? If any other cracking other than vibratory has occurred XI.M35 is not to be used. See recommended changes to add clarification.

XI.M35 – 3 In element 4 provide clarification of extent of opportunistic destructive testing to perform in event of a significant piping replacement that replaces or eliminates numerous welds.

XI.M35 – 4 In element 5 be clear that the cracking identified by One-Time Inspection or OE will undergo root cause analysis and that the need for periodic inspections going forward should only apply to the piping configurations that are determined to be susceptible to that cause. A certain piping loop may have vibration issues that other piping in the program are confirmed not to have.

**Program Description**

This program augments the existing American Society of Mechanical Engineers (ASME) Code, Section XI requirements and is applicable to small-bore ASME Code Class 1 piping and systems less than 4 inches nominal pipe size (NPS<4). The program includes pipes, fittings, branch connections, and all full and partial penetration (socket) welds. According to Table IWB-2500-1, Examination Category B-J Item No. B9.21 and B9.40 of the current ASME Code, a surface examination of small-bore Class 1 piping should be included for piping less than 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 of full-penetration welds, the inspection should be a volumetric examination. For a one-time inspection to detect cracking in socket welds, the inspection should be either a volumetric or opportunistic destructive examination. (Opportunistic destructive examination is performed when a weld is removed from service for other considerations, such as plant modifications. A sampling basis is used if more than 1 weld is removed) These are 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 cyclical (including thermal, mechanical, and vibration fatigue~~ory~~) loading or have performed effective design changes to mitigate cracking from vibration fatigue~~ory~~ loading. Should evidence of cracking of ASME Code Class 1 small-bore piping be revealed by a one-time inspection or previous operating experience, periodic inspection is proposed, as managed by a plant-specific AMP. If operating experience indicates previous design changes have not ~~been implemented to~~ effectively mitigated cracking from vibration~~ory~~ fatigue loading, periodic inspection is ~~also proposed~~ required, ~~as managed~~ by a plant-specific AMP.

**1.** ***Scope of Program:*** This program is a one-time inspection of a sample of ASME Code Class 1 piping less than NPS 4. 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 cyclical (including thermal, mechanical, and vibration fatigue~~ory~~) loading or have performed design changes that effectively mitigated~~ing~~ cracking from vibration fatigue~~ory~~ loading. ~~Should~~ If evidence of cracking of ASME Code Class 1 small-bore piping be revealed by a one-time inspection or previous operating experience, periodic inspection is proposed, as managed by a plant-specific AMP. If operating experience indicates design changes have not ~~been implemented to~~ effectively mitigated cracking from vibration fatigue~~ory~~ loading, periodic inspection is ~~also proposed~~required ~~, as managed~~ by a plant-specific AMP. This program includes measures to verify that degradation is not occurring; thereby either confirming that there is no need to manage age-related degradation or validating 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 Reports 1011955 (MRP-146) and 1018330 (MRP-146S).

**4.** ***Detection of Aging Effects:*** The inspection is designed to provide assurance, in plants that have not experienced cracking of ASME Code Class 1 small-bore piping due to stress corrosion or cyclical (including thermal, mechanical, and vibration fatigue ~~ory~~) loading, that aging of this piping is not occurring or that the effects of aging are not significant. For a one-time inspection to detect cracking in socket welds, the inspection should be either a volumetric or opportunistic destructive examination. (Opportunistic destructive examination is performed is removed when a weld is removed from service for other considerations, such as plant modifications. A sampling basis is used if more than 1 weld is removed) For a one-time inspection to detect cracking resulting from thermal and mechanical loading or intergranular stress corrosion of full penetration welds, the inspection should be a volumetric examination. Volumetric examination is performed using demonstrated techniques that are capable of detecting the aging effects.

**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 cyclical (including thermal, mechanical, and vibration fatigue ~~ory~~) loading is an issue. Evaluation of the inspection results may indicate the need for additional or periodic examinations (i.e., a plant-specific AMP for Class 1 small-bore piping using volumetric inspection methods consistent with ASME Code, Section XI, Subsection IWB). This inspection should be performed at a sufficient number of locations to ensure an adequate sample. This number, or sample size, is based on susceptibility, inspectability, dose considerations, operating experience, and limiting locations of the total population of ASME Code Class 1 small-bore piping locations.

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**XI.M36 External Surfaces Monitoring of Mechanical Components**

Comment/Basis:

XI.M36 – 1 Element 1 - Scope discusses cracking of stainless steel as and aging effect in polymeric components. Appears to be in wrong place. Also wrong aging effects are listed for polymers in some cases.

XI.M36 – 2 Element 4 - repeats requirements on qualifications and inspections that are not needed. Add requirements for the inspections of normally inaccessible and underground components to ensure they are appropriately identified when appropriate.

XI.M36 – 4 Element 6 - uses an acceptance criteria of "unchanged" for polymeric materials. Use of this criteria would result in rejection for even very minor changes in color, hardness and flexibility which are subjective examinations. Suggest using a criteria that any changes in these properties will be evaluated for continued service in the corrective action program to allow a proper and documented review of the condition.

1. ***Scope of Program:*** This program visually inspects the external surface of in-scope mechanical components and monitors external surfaces of metallic components in systems within the scope of license renewal and subject to AMR for loss of material and leakage. Cracking of stainless steel components exposed to an air environment containing halides may also be managed. This program also visually inspects and monitors the external surfaces of polymeric components in mechanical systems within the scope of license renewal and subject to AMR for change in material properties (such as hardening and loss of strength), cracking ~~of stainless steel components exposed to an air environment containing halides~~, and loss of material due to wear. This program manages the effects of aging of polymer materials in all environments to which these materials are exposed.

The program may also be credited with managing loss of material from internal surfaces of metallic components and with loss of material, cracking ~~hardening~~, and change in material properties from the internal surfaces of polymers, 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.

4. ***Detection of Aging Effects:*** This program manages aging effects of loss of material, cracking and change in material properties using visual inspection. For coated surfaces, confirmation of the integrity of the paint or coating is an effective method for managing the effects of corrosion on the metallic surface.

When required by the ASME Code, inspections are conducted in accordance with the applicable code requirements. In the absence of applicable code requirements, plant-specific visual inspections are performed of metallic and polymeric component surfaces using plant-specific personnel qualification procedures. The inspections are capable of detecting age-related degradation and are used at a frequency not to exceed one refueling cycle. This frequency accommodates inspections of components that may be in locations that are normally only accessible during outages. Surfaces that are not readily visible during plant operations and refueling outages or access is physically restricted (underground) are inspected when they are made accessible and at such intervals that would ensure the components intended function is maintained. The inspections of underground components shall be conducted during each 10 year period beginning 10 years prior to the period of extended operation. These normally inaccessible and underground components should be clearly identified in the program scope and inspection intervals provided. Surfaces that are insulated may be inspected when the external surface is exposed (i.e., during 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.

~~Plant-specific visual inspections are performed of flexible polymeric component surfaces using plant-specific personnel qualification procedures. The inspections are capable of detecting age-related degradation and are used at a frequency not to exceed one refueling cycle. Visual inspection of flexible polymeric components is performed using a plant-specific procedures by inspectors qualified through plant procedures whenever the component surface is accessible.~~ Visual inspection will identify indirect indicators of flexible polymer hardening and loss of strength and 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 should be 100 percent of accessible components. Visual inspection will identify direct indicators of loss of material due to wear to include dimensional change, scuffing, and for flexible polymeric materials with internal reinforcement, the exposure of reinforcing fibers, mesh, or underlying metal. Manual or physical manipulation can be used to augment visual inspection to confirm the absence of hardening and loss of strength for flexible polymeric materials (e.g., HVAC flexible connectors) where appropriate. The sample size for manipulation should be at least 10

**6. *Acceptance Criteria:*** For each component/aging effect combination, the acceptance criteria are defined to ensure that the need for corrective actions will be identified before loss of intended functions. For metallic surfaces, any indications of relevant degradation detected are evaluated. For stainless steel surfaces, a clean shiny surface is expected. The appearance of discoloration may indicate the loss of material on the stainless steel surface. For aluminum and copper alloys exposed to marine or industrial environments, any indications of relevant degradation that could impact their intended function are evaluated. For flexible polymers, a uniform surface texture and uniform color with no unanticipated dimensional change is expected. Any abnormal surface condition may be an indication of an aging effect for metals and for polymers. For flexible materials ~~to be considered acceptable~~, ~~the inspection results should indicate that the flexible polymer material has not~~ changes~~d~~ in~~ts~~ physical properties (e.g., the hardness, flexibility, physical dimensions, and color of the material are unchanged from when the material was new) should be evaluated for continued service in the corrective action program. Cracks should be absent within the material. For rigid polymers, surface changes affecting performance, such as erosion, cracking, crazing, checking, and chalking, are subject to further investigation. Acceptance criteria include design standards, procedural requirements, current licensing basis, industry codes or standards, and engineering evaluation.

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

Comment/Basis:

XI.M38 – 1 In the program description the last paragraph is not clear on the limitations of use of the program when failures have occurred. Recommend that section be reworded to clarify that repetitive failures would require use of a plant specific program.

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

**Program Description**

The program consists of inspections of the internal surfaces of metallic steel piping, piping components, ducting, polymeric components, and other components that are exposed to air indoor uncontrolled, air outdoor, condensation, and any water system not covered by other than open-cycle cooling water, treated water, and fire water 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 ensure that existing environmental conditions are not causing material degradation that could result in a loss of component intended functions. For certain materials, such as polymers, physical manipulation or pressurization (e.g., hydrotesting) to detect hardening or loss of strength should be used to augment the visual examinations conducted under this program. If visual inspection of internal surfaces is not possible, then the applicant needs to provide a plant-specific program.

This program is not intended for use on piping and ducts where repetitive failures have occurred from loss of material ~~from corrosion~~ and resulted~~ing~~ in loss of a system intended function. If operating experience indicates that there have been ~~on-going~~ repetitive failures caused by loss of material ~~due to corrosion~~, a plant specific program will be required. Following a failure, this program may be used if the failed material is replaced by one which is more corrosion resistant in the environment of interest.

## XI.M39 Lubricating Oil Analysis

Comment/Basis:

XI.M39 – 1 Delete references to SCC which has no basis if water is not present. This program ensures the lack of significant moisture that could cause cracking. No known OE exists of this occurring in an oil environment. Minor text changes also provided for clarification in that this program monitors for impurities.

### 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 to prevent or mitigate age-related degradation of components within the scope of this program. This 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.

Although primarily a sampling program, the lube oil analysis program is generally effective in monitoring and controlling impurities. This report identifies those circumstances in which the lube oil analysis program is to be augmented to manage the effects of aging for license renewal. Accordingly, in certain cases as identified in this report, verification of the effectiveness of the lube oil analysis program is undertaken to ensure that significant degradation is not occurring and that the component's intended function is maintained during the extended period of operation. As discussed in this report for these specific cases, an acceptable verification program is a one-time inspection of selected components at susceptible locations in the system. ~~Evidence of cracking is detected through the Chapter XI.M32, "One-Time Inspection" program.~~

1. ***Scope of Program:*** The program manages aging effects of loss of material due to corrosion or wear~~, cracking due to stress corrosion cracking (SCC)~~, or reduction of heat transfer due to

fouling. Components within the scope of the program include piping, piping components, and piping elements; heat exchanger tubes; reactor coolant pump elements; and any other plant components subject to aging management review that are exposed to an environment of lubricating oil (including non-water-based hydraulic oils).

**2. *Preventive Actions:*** The Lubricating Oil Analysis Program maintains oil system contaminants (primarily water and particulates) within acceptable limits.

***3. Parameters Monitored/Inspected:*** This program performs a check for water and a particle count to detect evidence of contamination by moisture or excessive corrosion. ~~Evidence of cracking is detected through the Chapter XI.M32, "One-Time Inspection" program.~~

**4. *Detection of Aging Effects:*** Moisture or corrosion products increase the potential for, or may be indicative of, loss of material due to corrosion, ~~cracking due to SCC (in stainless steel components at temperatures greater than 140oF),~~ or reduction of heat transfer due to fouling. The program performs periodic sampling and testing of lubricating oil for moisture and corrosion particles in accordance with industry standards. The program recommends sampling and testing of the old oil following periodic oil changes or on a schedule consistent with equipment manufacturer's recommendations or industry standards (e.g., American Society for Testing of Materials [ASTM] D 6224-02). Plant-specific operating experience also may be used to augment manufacturer's recommendations or industry standards in determining the schedule for periodic sampling and testing when justified by prior sampling results.

**10. *Operating Experience:*** The operating experience at some plants has identified (a) water in the ~~lubricating~~ oil and (b) particulate contamination. However, no instances of component failures attributed to lubricating oil contamination have been identified.

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## XI.M40 Monitoring of Neutron-Absorbing Materials Other Than Boraflex

Comment/Basis:

XI.M40 - 1 Element 4 - This program as written specifies an inspection frequency of 10 years minimum. License amendments that have approved the use of newer materials may specify different frequencies. To prevent conflicts in testing frequencies the option of following approved SER requirements should be allowed.

XI.M40 – 2 Element 3 - if Boral was to experience a loss of material, it would not result in shrinkage. Loss of material in Boral is conceptually similar to selective leaching, in that the B-10 would be selectively removed and the Boral sheet/coupon would simply become less dense without a change in dimension. Changes in dimensions are not typically shrinkage but increases in thickness as a result of the Al cladding separating from the inner Al-B alloy. The way it reads now, "...exposure to wet pool environment may cause shrinkage resulting in a loss of material..." is somewhat inaccurate for Boral. Suggest deleting shrinkage.

XI.M40 – 3 References - add IE Notice 2009-26, "Degradation of Neutron-Absorbing Materials in the Spent Fuel Pool"

**Program Description**

A monitoring program is implemented to assure that degradation of the neutron-absorbing material used in spent fuel pools that could compromise the criticality analysis will be detected. The applicable aging management program (AMP) relies on periodic inspection, testing, monitoring, and analysis of the criticality design to assure that the required 5 percent sub-criticality margin is maintained during the period of license renewal.

**Evaluation and Technical Basis**

**1. *Scope of Program*:** The AMP manages the effects of aging on neutron-absorbing components/materials used in spent fuel racks.

**3. Parameters Monitored/Inspected:** For these materials, gamma irradiation and/or long-term exposure to the wet pool environment may cause ~~shrinkage resulting in~~ loss of material, and changes in dimension (such as gap formation, formation of blisters, pits and bulges) that could result in loss of neutron-absorbing capability of the material. The parameters monitored include the physical condition of the neutron-absorbing materials, such as in-situ gap formation, geometric changes in the material (formation of blisters, pits, and bulges) as observed from coupons or in situ, and decreased boron areal density, etc. The parameters monitored is directly related to determination of the loss of material or loss of neutron absorption capability of the material(s).

**4. *Detection of Aging Effects*:** The loss of material and the degradation of the neutron absorbing material capacity are determined through coupon and/or direct in-situ testing. Such testing should include periodic verification of boron loss through areal density measurement of coupons or through direct in situ techniques which may include measurement of boron areal density, geometric changes in the material (blistering, pitting, and bulging), and detection of gaps through blackness testing. The frequency of the inspection and testing depends on the condition of the neutron-absorbing material and is determined and justified with plant-specific operating experience by the licensee, not to exceed 10 years. <u>For materials reviewed and approved for use by license amendment, testing and inspection should follow SER requirements.</u>

**References**

NRC Information Notice 2009-26, *Degradation of Neutron-Absorbing Materials in the Spent Fuel Pool, U.S. Regulatory Commission*, October 28, 2009

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## SRP Section 4.2 – Metal Fatigue Analysis

| Section / Page No. | Comment |
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| **4.2.2.1.1.2** | In several places, ft-lb were changed to joules, such as replacing "50 ft-lb" with "68 joules (50 ft-lb)". As ft-lb is more common usage than joules, shouldn't this say "50 ft-lb (68 joules)"? This would then be consistent with the format in 10 CFR 50, Appendix G. |
| **4.2.2.1.4.1** | Minor wording changes only.<br>**10 CFR 54.21(c)(1)(iii)** An applicant for renewal of a license should address this issue by noting that it will be handled through a re-application under 10 CFR 50.55a(a)(3). An applicant for ~~a~~ license ~~to operate such~~ renewal of a BWR may provide justification to extend this relief into the period of extended operation in accordance with BWRVIP-74-A (Ref 8), which is the revised and NRC approved version of BWRVIP-74 (Ref. 9). The staff's review of BWRVIP-74 (Ref. 9) is contained in an October 18, 2001 letter to C.Terry, BWRVIP Chairman (Ref. 10). ~~Section A.4.5 of Report BWRVIP-74-A indicates that~~ Appendix E of the staff's final safety evaluation report (FSER) (Ref. 10) conservatively evaluated BWR RPV's to ~~have~~ 64 effective full power years (EFPY), which is 10 EFPY greater than the maximum of what is realistically expected for the end of the license renewal period. Since this is a generic analysis, a licensee relying on BWRVIP-74-A should provide plant-specific information to demonstrate that at the end of the renewal period, the circumferential beltline weld materials meet the limiting conditional failure probability for circumferential welds specified in Appendix E of the FSER (Ref. 10) and that operator training and procedures are utilized during the |

| Section / Page No. | Comment |
|---|---|
| | license renewal term to limit the frequency for cold over-pressure events to the amount specified in the NRC FSER (Ref. 10). |
| **4.2.3.1.1** | Put the discussion of neutron fluence after 4.2.3.1.1 and before 4.2.1.1.1/4.2.3.1.1.2/4.2.3.1.1.3 as it applies to all three subsections. |
| **4.2.3.1.1.2** | The second half of this section is confusing. Many plants have surveillance data that shows a larger drop in USE than that predicted by RG 1.99 Revision 2, Position 1. RG 1.99 Position 2 allows the adjustment of the predicted USE based on the plant's surveillance data. Only plants without adequate surveillance data to make the adjustment could have larger reductions in USE than that predicted by RG 1.99. In the BWRVIP Integrated Surveillance Program, there is no BWR in this condition. Suggest rewording the last part of the paragraph as shown below.<br><br>*For Boiling Water Reactors, the staff confirms that the beltline materials are evaluated in accordance with Renewal Applicant Action Item 10 in the staff's SER, for BWRVIP-74 (Letter to C. Terry dated October 18, 2001) (Ref. 10). Action Item 10: To demonstrate that the beltline materials meet the Charpy USE criteria specified in Appendix B of BWRVIP-74-A or the NRC FSER (Ref. 10), the applicant demonstrates that the projected percent reduction in Charpy USE for their beltline materials is less than that specified for the limiting BWR/3-6 plates and the limiting non-Linde 80 submerged arc welds. ~~and that the percent reduction in Charpy USE for their surveillance weld and plate are less than or equal to the values projected using the methodology in NRC RG 1.99, Revision 2.~~*<br><br>If there should be a BWR with more embrittlement than RG 1.99 Position 1 and not enough credible surveillance data to correct it (say someone drops out of the ISP for some reason); the real question is what action would the staff require in this case? |
| **4.2.3.1.3** | Put the discussion of neutron fluence after 4.2.3.1.3 and before 4.2.1.3.1/4.3.2.1.4.2/4.2.1.3.3 as it applies to all three subsections.<br><br>Does every plant use ¾ T to determine P-T limits? If not, this should just say ¼ T. |
| **4.2.3.1.3.1** | The ~~documented results of the~~ projected neutron fluence for the ¼T and ¾ T locations at the end of the period of extended operation ~~they~~ are bounded by the neutron fluences used to develop the existing P-T limit |

| Section / Page No. | Comment |
|---|---|
| | analysis. |

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## SRP Section 4.3 – Metal Fatigue Analysis

| Section / Page No. | Comment |
|---|---|
| **4.3.2** | Subpart 2 describes the stress reduction factors as "maximum allowable". However, there are not min/max values for the factors listed in Table 4.3-1. Consider clarifying as shown below.<br><br>2. Implicit fatigue-based maximum allowable stress calculations for piping components designed to USAS ANSI B31.1 (Ref. 2) requirements, and ASME Code Class 2 and 3 components designed to ASME Section III design requirements that are similar to the guidance in ANSI B31.1.<br><br>ANSI B31.1 applies only to piping and does not call for an explicit fatigue analysis. It specifies allowable stress levels based on the number of anticipated full thermal range transient cycles. The specific ~~maximum allowable~~ stress range reduction~~s~~ factors due to full thermal cycles are listed in Table 4.3-1. |
| **4.3.1** | Most metallurgists and fatigue experts would disagree that a CUF of 1.0 assumes there is a crack. Suggested rewording below.<br><br>. . . A CUF below~~above~~ a value of one provides assurance that no ~~assumes that a small, but analyzable~~ crack has been formed. A CUF above a value of one allows for the possibility that a crack may form, and that if~~If undetected or~~ left untreated, the crack could~~will~~ propagate exponentially under fatigue loading and eventually lead to coolant leakage in reactor pressure boundary components, or even general structural failure. |

| Section / Page No. | Comment |
|---|---|
| **4.3.2.1.1.3** | Don't limit GALL X.M1 to the RCS pressure boundary. There are lots of non-RCS pressure boundary items (containment buildings, supports, PWR secondary side items, etc.) that are identified in GALL Sections II through VIII that have cycles that need monitoring with this program. SRP-LR Section 4.3.1 acknowledges this. |
| **4.3.2.1.3** | The last paragraph is not clear. It seems to imply that the nickel alloy Fens from RG 1.207 can be used with the existing ASME fatigue curves. This is not what RG 1.207 (or NUREG\CR-6909 says). This paragraph should be clarified. Note that 4.3.3.1.3 does not allow this. |
| **4.3.2.1.4** | There is no 4.3.2.1.4, the report goes from 4.3.2.1.3 to 4.3.2.1.5. |
| **4.3.2.1.5.3** | Why isn't GALL X.M1 an acceptable basis for accepting flaw growth and fracture mechanics analyses? Seems like an explanation is in order. Note that section 4.3.3.1.5.3 re-iterates this statement while section 4.3.3.1.1.3 seems to contradict it. |
| **4.3.3.1.1.2** | "Adequate" needs definition. Suggest rewording to eliminate it.<br>The operating transient experience is reviewed to ensure that~~and a list of~~ the increased number of ~~assumed~~ transients used for any re-analysis meets or exceeds the number of transients projected to the end of the period of extended operation ~~are reviewed to ensure that the transient projection is adequate~~. The revised CUF calculations ~~based on the projected number of assumed transients~~ are reviewed to ensure that the CUF remains less than or equal to one at the end of the period of extended operation. For consistency purposes, the review also includes an assessment of the TLAA information against relevant design basis information and CLB information (including applicable cycle-counting requirements in technical specifications). |

## XI.S1 ASME SECTION XI, SUBSECTION IWE

XI.S1 – 1 Program Description: 1) GL 98-04 and XI.S1 ASME Section XI, Subsection IWE are appropriate for maintaining and monitoring coatings inside containment. 2) This increase in scope which in Element 4 requires " surface examination, in addition to visual examination to detect cracking .." is not supported by two OEs cited; one OE on bellows addresses stainless steel cracking as result of contamination and the other OE addresses which is torus cracking apparently due to a design issue "lack of an HPCI turbine exhaust pipe sparger". The industry OE does not show cracking of penetration sleeves, associated steel components and bellows due to cyclic loads to be a problem which would warrant an augmented requirement for a supplemental surface examination. The proposed new requirement is above and beyond the requirement of ASME Code Section XI, IWE and 10CFR50.55a. We believe ASME Section XI-IWE and 10CFR 50, Appendix J provide adequate requirement for inspection of penetration components.

We recommend that this new requirement be eliminated or at the most limited to bellows for cyclic loads and possibly stainless steel and dissimilar metal welds for SCC. Carbon steel penetration sleeves, closures and flued heads should not be subject to augmented supplemental surface examinations.

XI.S1 – 2 Element 1: GL 98-04 and XI.S1 ASME Section XI, Subsection IWE are appropriate for maintaining and monitoring coatings inside containment.

XI.S1 – 3 Element 2: The above EPRI and NUREG bolting recommendations as listed, appear to be design and installation and maintenance procedural recommendations more than Aging Management Program changes. A change of bolting material, torque, tension or lubricant selections for a specific application would likely require a design change. OE does not show degradation and failure of IWE bolting, except for the limited specific OE on large HS bolts of specific limited application bolting material for NSSS IWE supports. Other changes do not appear necessary or warranted by industry experience and OE.

XI.S1 – 4 Element 4: This increase in scope which requires " surface examination, in addition to visual examination to detect cracking .." is not supported by two OEs cited; one OE on bellows addresses stainless steel cracking as result of contamination and the other OE addresses which is torus cracking apparently due to a design issue "lack of an HPCI turbine exhaust pipe sparger". The industry OE does not show cracking of penetration sleeves, associated steel components and bellows due to cyclic loads to be a problem which would warrant an augmented requirement for a supplemental surface examination. The proposed new requirement is above and beyond the requirement of ASME Code Section XI, IWE and 10CFR50.55a. We believe ASME Section XI-IWE and 10CFR 50, Appendix J provide adequate requirement for inspection of penetration components.

We recommend that this new requirement be removed or at the most limited to bellows for cyclic loads and possibly stainless steel and dissimilar metal welds for SCC. Carbon steel penetration sleeves, closures and flued heads should not be subject to augmented supplemental surface examinations.

XI.S1 – 5 Element 10: This discussion is not relevant to this program.

XI.S1 – 6 References: These references do not apply to this program.

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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 and moisture barriers, and pressure-retaining bolting. This evaluation covers the 2004 edition[1], 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.

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 to detect aging effects. IWE specifies acceptance criteria, corrective actions, and expansion of the inspection scope when degradation exceeding the acceptance criteria is found.

~~Subsection IWE addresses coatings that prevent corrosion. XI.S8 is a protective coating monitoring and maintenance program that is recommended to ensure Emergency Core Cooling System (ECCS) operability, whether or not the coatings are credited in XI.S1~~.

The program attributes are augmented to incorporate aging management activities, recommended in the Final Interim Staff Guidance LR-ISG-2006-01, needed to address the potential loss of material due to corrosion in the inaccessible areas of the boiling water reactor (BWR) Mark I steel containment. ~~The attributes also are augmented to require surface examination for detection of cracking described in NRC Information Notice (IN) 92-20 and address recommendations delineated in NUREG-1339 and industry recommendations delineated in the Electric Power Research Institute (EPRI) NP-5769, NP-5067, and TR-104213 for structural bolting~~

### Evaluation and Technical Basis

**1. *Scope of Program:*** The scope of this program addresses the components of steel containments and steel liners of concrete containments specified in Subsection IWE-1000 as augmented by LR-ISG-2006-01. 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 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 addresses coatings that prevent~~

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[1] Refer to the GALL Report, Chapter I, for applicability of other editions of the ASME Code, Section XI.

~~corrosion. XI.S8 is a protective coating monitoring and maintenance program that is recommended to ensure ECCS operability, whether or not the coatings are credited in XI.S1.~~.

Subsection IWE exempts the following from examination: Components that are outside the boundaries of the containment as defined in the plant-specific design specification;

(b) Embedded or inaccessible portions of containment components that met the requirements of the original construction code of record;

(c) Components that become embedded or inaccessible as a result of containment structure (i.e., steel containments [Class MC] and steel liners for concrete containments [Class CC]) repair or replacement, provided IWE-1232 and IWE-5220 are met; and

(d) 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.

**2. *Preventive Action:*** The ASME Code Section XI, Subsection IWE, is a condition monitoring program. The program is augmented to include preventive actions that ensure moisture levels associated with an accelerated corrosion rate do not exist in the exterior portion of the BWR Mark I steel containment drywell shell. The actions consist of ensuring that the sand pocket area drains and/or the refueling seal drains are clear. ~~The program is also augmented to require that the selection of bolting material installation torque or tension and the use of lubricants and sealants are in accordance with the guidelines of EPRI NP-5769, EPRI TR-104213, and the additional recommendations of NUREG-1339 to prevent or mitigate degradation and failure of structural bolting.~~

**3. *Parameters Monitored or Inspected:*** Table IWE-2500-1 references the applicable section in IWE-2300 and IWE-3500 that identify the parameters examined or monitored. Non-coated surfaces are examined for evidence of cracking, discoloration, wear, pitting, excessive corrosion, arc strikes, gouges, surface discontinuities, dents, and other signs of surface irregularities. Painted or coated surfaces are examined for evidence of flaking, blistering, peeling, discoloration, and other signs of distress. Stainless steel penetration sleeves, dissimilar metal welds, bellows, and steel components that are subject to cyclic loading but have no current licensing basis fatigue analysis are monitored for cracking. The moisture barriers are examined for wear, damage, erosion, tear, surface cracks, or other defects that permit intrusion of moisture against inaccessible areas of the pressure retaining surfaces of the metal containment shell or liner. Pressure-retaining bolting is examined for loosening and material conditions that cause the bolted connection to affect either containment leak-tight or structural integrity.

As recommended in LR-ISG-2006-01, license renewal applicants with BWR Mark I containments should monitor the sand pocket area drains and/or the refueling seal drains for water leakage. The licensees should ensure the drains are clear to prevent moisture levels associated with accelerated corrosion rates in the exterior portion of the drywell shell.

**4. *Detection of Aging Effects:*** The examination methods, frequency, and scope of examination specified in 10 CFR 50.55a and Subsection IWE ensure that aging effects are detected before they compromise the design-basis requirements. IWE-2500-1 and the requirements of 10 CFR 50.55a provide information regarding the examination categories, parts examined, and examination methods to be used to detect aging.

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 percent 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 percent at the end of that interval.

After 40 years of operation, any future examinations are 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 percent 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 was 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 as specified in IWE-2500-1 and the requirements of 10 CFR 50.55a. The acceptability of inaccessible areas of the BWR Mark I steel containment drywell is evaluated when conditions exist in the adjacent accessible areas that could indicate the presence of or could result in degradation to such inaccessible areas. 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.

~~The requirements of ASME Section XI, Subsection IWE and 10 CFR 50.55a are augmented to require surface examination, in addition to visual examination, to detect cracking in stainless steel penetration sleeves, dissimilar metal welds, bellows, and steel components that are subject to cyclic loading but have no current licensing basis fatigue analysis~~

**5. *Monitoring and Trending:*** With the exception of inaccessible areas, all surfaces are monitored by virtue of the examination requirements on a scheduled basis. IWE-2420 specifies that:

(a) The sequence of component examinations established during the first inspection interval shall be repeated.
(b) When examination results require evaluation of flaws or areas of degradation in accordance with IWE-3000, and 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 IWE-2500-1 and the regular inspection schedule is continued.

Applicants for license renewal for plants with BWR Mark I containment should augment IWE monitoring and trending requirements to address inaccessible areas of the drywell. The applicant should consider the following recommended actions based on plant-specific operating experience.

(a) Develop a corrosion rate that can be inferred from past ultrasonic testing (UT) examinations or establish a *corrosion rate using representative samples in similar operating conditions*, materials, and environments. If degradation has occurred, provide a technical basis using the developed or established corrosion rate to demonstrate that the drywell shell will have sufficient wall thickness to perform its intended function through the period of extended operation.

(b) Demonstrate that UT measurements performed in response to U.S. Nuclear Regulatory Commission (NRC) Generic Letter (GL) 87-05 did not show degradation inconsistent with the developed or established corrosion rate.

**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 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. For some examinations, such as augmented examinations, numerical values are specified for the acceptance standards. For the containment steel shell or liner, material loss locally exceeding 10 percent of the nominal containment wall thickness or material loss that is projected to locally exceed 10 percent of the nominal containment wall thickness before the next examination are documented. Such areas are corrected by repair or replacement in accordance with IWE-3122 or accepted by engineering evaluation. Cracking of stainless steel penetration sleeves, dissimilar metal welds, bellows, and steel components that are subject to cyclic loading but have no current licensing basis fatigue analysis is corrected by repair or replacement or accepted by engineering evaluation.

**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 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. 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.

If moisture has been detected or suspected in the inaccessible area on the exterior of the Mark I containment drywell shell or the source of moisture cannot be determined subsequent to root cause analysis then:

(a) Include in the scope of license renewal any components that are identified as a source of moisture, if applicable, such as the refueling seal or cracks in the stainless liners of the refueling cavity pools walls, and perform aging management review.
(b) Identify surfaces requiring examination by implementing augmented inspections for the period of extended operation in accordance with Subsection IWE-1240, as identified in Table IWE-2500-1, Examination Category E-C.
(c) Use examination methods that are in accordance with Subsection IWE-2500.
(d) Demonstrate, through use of augmented inspections performed in accordance with Subsection IWE, that corrosion is not occurring or that corrosion is progressing so slowly that the age-related degradation will not jeopardize the intended function of the drywell shell through the period of extended operation.

**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 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.

**9. *Administrative Controls:*** IWA-6000 provides specifications for the preparation, submittal, and retention of records and reports. As discussed in the appendix to this report, the staff finds the requirements of 10 CFR Part 50, Appendix B, acceptable to address administrative controls.

**10. *Operating Experience:*** ASME Section XI, Subsection IWE, was incorporated into 10 CFR 50.55a in 1996. Prior to this time, operating experience pertaining to degradation of steel components of containment was gained through the inspections required by 10 CFR Part 50, Appendix J and ad hoc inspections conducted by licensees and the NRC. NRC Information Notice (IN) 86-99, IN 88-82, IN 89-79, IN 2004-09, and NUREG-1522 described occurrences of corrosion in steel containment shells. NRC GL 87-05 addressed the potential for corrosion of BWR Mark I steel drywells in the "sand pocket region." NRC IN 97-10 identified specific locations where concrete containments are susceptible to liner plate corrosion; IN 92-20 described an instance of containment bellows cracking, resulting in loss of leak tightness. More recently, IN 2006-01 described a through-wall cracking and its probable cause in the torus of a BWR Mark I containment. The cracking was identified by the licensee in the heat-affected zone at the high pressure cooling injection (HPCI) turbine exhaust pipe torus penetration. The licensee concluded that the cracking was most likely initiated by cyclic loading due to condensation oscillation during HPCI operation. These condensation oscillations induced on the torus shell may have been excessive due to a lack of an HPCI turbine exhaust pipe sparger that many licensees have installed. Other operating experience indicates that foreign objects embedded in concrete have caused through-wall corrosion of the liner plate at a few plants with reinforced concrete containments. The program is to consider the liner plate and containment shell corrosion and cracking concerns described in these generic communications. Implementation of the ISI requirements of

Subsection IWE, in accordance with 10 CFR 50.55a, augmented to consider operating experience, and as recommended in LR-ISG-2006-01, is a necessary element of aging management for steel components of steel and concrete containments through the period of extended operation.

~~Degradation of threaded bolting and fasteners in closures for the reactor coolant pressure boundary has occurred from boric acid corrosion, stress corrosion cracking (SCC), and fatigue loading (NRC IE Bulletin 82-02, NRC GL 91-17). SCC has occurred in high strength bolts used for nuclear steam supply system component supports (EPRI NP-5769). The augmented ASME Section XI, Subsection IWE, incorporating recommendations documented in EPRI NP-5769 and TR-104213, is necessary to ensure containment bolting integrity.~~

**References**

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ASME Section XI, *Rules for Inservice Inspection of Nuclear Power Plant Components,* Subsection IWL, *Requirements for Class CC Concrete Components of Light-Water Cooled Power Plants,* The ASME Boiler and Pressure Vessel Code, 2004 edition as approved in 10 CFR 50.55a, The American Society of Mechanical Engineers, New York, NY. April 2010 DRAFT XI S1-7 NUREG-1801, Rev. 2

~~EPRI NP-5067, Good Bolting Practices, A Reference Manual for Nuclear Power Plant Maintenance Personnel, Volume 1: Large Bolt Manual, 1987; Volume 2: Small Bolts and Threaded Fasteners, Electric Power Research Institute, 1990.~~

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

~~EPRI TR-104213, Bolted Joint Maintenance & Application Guide, Electric Power Research Institute, December 1995.~~

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

NRC Generic Letter 87-05, *Request for Additional Information Assessment of Licensee Measures to Mitigate and/or Identify Potential Degradation of Mark I Drywells,* U.S. Nuclear Regulatory Commission, March 12, 1987.

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

NRC Information Notice 86-99, *Degradation of Steel Containments,* U.S. Nuclear Regulatory Commission, December 8, 1986 and Supplement 1, February 14, 1991.

NRC Information Notice 88-82, *Torus Shells with Corrosion and Degraded Coatings in BWR Containments,* U.S. Nuclear Regulatory Commission, October 14, 1988 and Supplement 1, May 2, 1989.

NRC Information Notice 89-79, *Degraded Coatings and Corrosion of Steel Containment Vessels,* U.S. Nuclear Regulatory Commission, December 1, 1989 and Supplement 1, June 29, 1989.

NRC Information Notice 92-20, *Inadequate Local Leak Rate Testing,* U.S. Nuclear Regulatory Commission, March 3, 1992.

NRC Information Notice 97-10, *Liner Plate Corrosion in Concrete Containment,* U.S. Nuclear Regulatory Commission, March 13, 1997.

NRC Information Notice 2004-09, *Corrosion of Steel Containment and Containment Liner*, U.S. Nuclear Regulatory Commission, April 27, 2004.

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## XI.S3 ASME SECTION XI, SUBSECTION IWF

XI.S3 – 1 Element 2: This program is a condition monitoring program and should not provide preventive actions.

XI.S3 – 2 Element 3: Not all high strength bolts are susceptible to SCC.

XI.S3 – 3 Element 4: Cracking of high strength bolts is not supported by the OE cited for structural bolts. Existing OE is for certain material, size, torque and lubricant applications only and should not be generically applied to all high strength bolts of 150 ksi or more. The Operating Experience cited in NUREG 1801 stated "SCC has occurred in high strength bolts used for nuclear steam supply system component supports (EPRI NP-5769)". The above is cited as operating experience in XI.M18, XI.S1, XI.S3, XI.S6, and XI.S7. The OE cited in NUREG 1801 is NP-5769 (issued in 1988) and was found only in certain specific materials and specifically for NSSS component supports and should not be generically applied to all structural high strength bolts. In certain cases the failures noted were attributed contributing factors including use of molybdenum disulfide thread lubricant which is considered a corrosive environment and is not used in structural steel installations. While EPRI NP-5769 Volume 1, Table 11-1 does list A490 bolts for ductile failures and failure due improper torque, and one instance of a special 4140 material with 200 ksi minimum yield where the A490 specification was used for heat treatment requirements (not an A490 structural bolt) where SCC was noted and associated with a high preload and borated water environment. No SCC failures were noted for commonly used materials in Structural Steel bolting including A307, A325 or A490 bolts in a structural steel application.

Industry documents including NUREG-1339 have not identified any determination specifically as to the ASTM A-490 material's susceptibility to SCC, but rather a determination of the yield stress level at which generic materials should be considered for the possibility of SCC vulnerability. In order for SCC to occur in high strength bolting, three parameters must exist; (1) a corrosive environment, (2) a susceptible material, and (3) high sustained tensile stresses. The absence of any one of these three parameters eliminates the material's susceptibility to SCC. High Strength Structural Bolts including A490 bolts are not subject to high-sustained preload stress and lubricants containing contaminants, such as $MoS_2$, are not approved for use. Therefore, stress corrosion cracking is not considered an applicable aging effect requiring management. The structural bolting wil continue to be visually inspected for loss of preload due to self loosening and loss of material due to corrosion and also visually monitor the associated structural steel and connections for loss of material or other adverse conditions.

The high strength bolts used for structural applications are A325 or A490 Bolts. A325 bolts have a minimum yield of 92 ksi (unlikely to have an actual yield of 150 ksi or more), and A490 bolts have a minimum yield of 130 ksi (potential that some A490 may reach actual yield of 150 ksi or more). Most structural bolts are 1" or less in diameter, however bolts in large girders or supporting large equipment may be over 1" diameter. Test reports are not generally traceable to installation locations.

The AISC "Guide to Design for Bolted and Riveted Joints" Second Edition published in 2001 addresses this concern in section 4.8 and concludes that the tests indicate that black (not galvanized) A490 bolts can be used without problems from "brittle" failures (failures due to hydrogen stress cracking or stress corrosion cracking) in most environments. It was concluded that galvanized A490 bolts should not be used in structures. This same section also concluded that black and galvanized A325 bolts behave satisfactorily with regard to hydrogen stress and stress corrosion cracking in most corrosive environments. AISC publications do not recommend or require volumetric or surface examinations of installed structural bolts for stress corrosion cracking.

This new requirement is above and beyond the requirement of ASME Code Section XI, IWF. We recommend that this new requirement be removed or at the most limited to the specific bolting material types identified in the OE cited on NSSS supports. Volumetric or surface examinations for cracking of high strength bolts should not be generically imposed for all bolts with actual yield strength of 150 ksi or more and greater than 1" diameter. Structural high strength bolts, including A325 and A490 bolts do not have a history of stress corrosion cracking in most environments.

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## XI.S3 ASME SECTION XI, SUBSECTION IWF

### Program Description

10 CFR 50.55a imposes the inservice inspection requirements of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, Section XI, for Class 1, 2, 3, and metal containment (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 2004 edition[2] of the ASME Code 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 component 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 requirements of subsection IWF are augmented to include monitoring of high-strength structural bolting (actual measured yield strength greater than or equal to 150 ksi or 1,034 MPa) for cracking. The program is augmented to incorporate recommendations delineated in NUREG-1339 and industry recommendations delineated in the Electric Power Research Institute (EPRI) NP-5769, NP-5067, and TR-104213 for high-strength structural bolting. These recommendations emphasize proper selection of bolting material, lubricants, and installation

[2] Refer to the GALL Report, Chapter I, for applicability of other editions of the ASME Code, Section XI.

torque or tension to prevent or minimize loss of bolting preload and cracking of high-strength bolting.

**Evaluation and Technical Basis**

1. ***Scope of Program:*** This program addresses supports for ASME Class 1, 2, and 3 piping and components supports that are not exempt from examination in accordance with IWF -1230 and MC supports. The scope of the program includes support members, structural bolting, high strength structural bolting, support anchorage to the building structure, accessible sliding surfaces, constant and variable load spring hangers, guides, stops, and vibration isolation elements.

2. ***Preventive Action:*** No preventive actions are specified; Subsection IWF is an inspection program.~~Selection of bolting material and the use of lubricants and sealants is in accordance with the guidelines of EPRI NP-5769, EPRI TR-104213, and the additional recommendations of NUREG-1339 to prevent or mitigate degradation and failure of safety-related bolting. Operating experience and laboratory examinations show that the use of molybdenum disulfide (MoS2) as a lubricant is a potential contributor to stress corrosion cracking (SCC), especially when applied to high strength bolting. Thus, molybdenum disulfide and other lubricants containing sulfur should not be used. Preventive measures also include using bolting material that has an actual measured yield strength less than 150 ksi or 1,034 MPa. Structural bolting replacement and maintenance activities include appropriate preload and proper tightening (torque or tension) as recommended in EPRI documents, American Society for Testing of Materials (ASTM) standards, American Institute of Steel Construction (AISC) Specifications, as applicable.~~

3. ***Parameters Monitored or Inspected***: The parameters monitored or inspected include corrosion; deformation; misalignment of supports; missing, detached, or loosened support items; improper clearances of guides and stops; and improper hot or cold settings of spring supports and constant load supports. Accessible areas of sliding surfaces are monitored for debris, dirt, or indications of excessive loss of material due to wear that could prevent or restrict sliding as intended in the design basis of the support. Elastomeric vibration isolation elements are monitored for cracking, loss of material, and hardening. Structural bolts are monitored for corrosion and loss of integrity of bolted connections due to self loosening and material conditions that can affect structural integrity. High-strength structural bolting (actual measured yield strength greater than or equal to 150 ksi or 1,034 MPa) susceptible to SCC should be monitored for SCC.

4. ***Detection of Aging Effects:*** The program requires that a sample of ASME Class 1, 2, and 3 component supports that are not exempt from examination and 100 percent of MC component supports be examined as specified in Table IWF-2500-1. The sample size examined for ASME Class 1, 2, and 3 component supports is as specified in Table IWF-2500-1. The extent, frequency, and examination methods are designed to detect, evaluate, or repair age-related degradation before there is a loss of component support intended function. The VT-3 examination method specified by the program can reveal loss of material due to corrosion and wear, verification of clearances, settings, physical displacements, loose or missing parts, debris or dirt in accessible areas of the sliding surfaces, or loss of integrity at bolted connections. The VT-3 examination can also detect loss of material and cracking of elastomeric vibration isolation elements. VT-3 examination of elastomeric vibration isolation elements should be supplemented by feel to detect hardening if the vibration isolation function is suspect. IWF-3200 specifies that visual examinations that detect surface flaws which exceed acceptance criteria may be

supplemented by either surface or volumetric examinations to determine the character of the flaw.

~~For high strength structural bolting (actual measured yield strength greater than or equal to 150 ksi or 1,034 MPa) in sizes greater than 1 inch nominal diameter, volumetric examination comparable to that of ASME Code Section XI, Table IWB-2500-1, Examination Category B-G-1 should be performed to detect cracking in addition to the VT-3 examination.~~This volumetric examination may be waived with adequate plant-specific justification.

**5. *Monitoring and Trending:*** The ASME Class 1, 2, 3, and MC component supports are examined periodically as specified in Table IWF-2500-1. As required by IWF-2420(a), the sequence of component support examinations established during the first inspection interval is repeated during each successive inspection interval, to the extent practical. Component supports whose examinations do not reveal unacceptable degradations are accepted for continued service. Verified changes of conditions from prior examination are recorded in accordance with IWA-6230. Component supports whose examinations reveal unacceptable conditions and are accepted for continued service by corrective measures or repair/ replacement activity are reexamined during the next inspection period. When the reexamined component support no longer requires additional corrective measures during the next inspection period, the inspection schedule may revert to its regularly scheduled inspection. Examinations that reveal indications which exceed the acceptance standards and require corrective measures are extended to include additional examinations in accordance with IWF-2430.

**6. *Acceptance Criteria:*** The acceptance standards for visual examination are specified in IWF-3400. IWF-3410(a) identifies the following conditions as unacceptable:

(a) Deformations or structural degradations of fasteners, springs, clamps, or other support items;
(b) Missing, detached, or loosened support items, including bolts and nuts;
(c) Arc strikes, weld spatter, paint, scoring, roughness, or general corrosion on close tolerance machined or sliding surfaces;
(d) Improper hot or cold positions of spring supports and constant load supports;
(e) Misalignment of supports; and
(f) Improper clearances of guides and stops.

Other unacceptable conditions include,
(a) Loss of material due to corrosion or wear, which reduces the load bearing capacity of the component support;
(b) Debris, dirt, or excessive wear that could prevent or restrict sliding of the sliding surfaces as intended in the design basis of the support;
(c) Cracked or sheared bolts, including high strength bolts, and anchors; and
(d) Loss of material, cracking, and hardening of elastomeric vibration isolation elements that could reduce the vibration isolation function.

The above conditions may be accepted provided the technical basis for their acceptance is documented.

**7. *Corrective Actions:*** Identification of unacceptable conditions triggers an expansion of the inspection scope, in accordance with IWF-2430, and reexamination of the supports requiring corrective actions during the next inspection period, in accordance with IWF-2420(b). In accordance with IWF-3122, supports containing unacceptable conditions are evaluated or tested or corrected before returning to service. Corrective actions are delineated in IWF-3122.2.

IWF-3122.3 provides an alternative for evaluation or testing to substantiate structural integrity and/or functionality. 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:*** To date, IWF sampling inspections have been effective in managing aging effects for ASME Class 1, 2, 3, and MC supports. There is reasonable assurance that the Subsection IWF inspection program will be effective in managing the aging of the in-scope component supports through the period of extended operation.

Degradation of threaded bolting and fasteners has occurred from boric acid corrosion, SCC, and fatigue loading (NRC IE Bulletin 82-02, NRC Generic Letter 91-17). SCC has occurred in high strength bolts used for NSSS component supports (EPRI NP-5769).

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## XI.S5 MASONRY WALLS

XI.S5-1 Comment element 4: Industry guidance found in NUREG-1522, NEI 96-03, and ACI 349 recommend a 5 to 10 year frequency. Program inspection frequency should be addressed as a part of Maintenance Rule implementation. Industry guidance NUREG-1522, "Assessment of Inservice Conditions of Safety-Related Nuclear Plant Structures, and NEI 96-03, "INDUSTRY GUIDELINE FOR MONITORING THE CONDITION OF STRUCTURES AT NUCLEAR POWER PLANTS" recommends a 5 to 10 year frequency. ACI 349 provides guidance for inspecting some structures at greater than a 5 year frequency.

Plants have established their maintenance rule program including inspection frequency based on Maintenance Rule in accordance with 10CFR 50.65, "Requirements for monitoring the effectiveness of maintenance at nuclear power plants". 10CFR 50.65, "MR", does not require 5 year inspection frequency for all the structures. The frequencies of the inspections are established based on risk-informed evaluation process relative to their significant to public health and safety. As such, some structures have inspection frequency between 5 to 10 years. And, some structures based in their site specific OE are inspected even more frequent than every 5 years. Some structures are already in 5 year inspection frequency. For example, Inspection of Water Control Structures governed by RG 1.127 are already conducted on a 5 year or more frequently if conditions warrant as required by RG 1.127. Concrete Containment IWL inspections are also generally performed at a 5 year frequency.

We recommend that this new requirement be removed. Imposing 5 year inspection frequency to all structures and masonry walls seems to be extreme measure without any bases. The inspection frequency should be established based on service and condition assessment of each structure in accordance with Maintenance Rule 10CFR 50.65. That could vary from 5 to 10 years.

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## XI.S5 MASONRY WALLS

### Program Description

Nuclear Regulatory Commission (NRC) IE Bulletin (IEB) 80-11, "Masonry Wall Design," and NRC Information Notice (IN) 87-67, "Lessons Learned from Regional Inspections of Licensee Actions in Response to IE Bulletin 80-11," constitute an acceptable basis for a masonry wall aging management program (AMP). IEB 80-11 required (a) the identification of masonry walls in close proximity to or having attachments from safety-related systems or components and (b) the evaluation of design adequacy and construction practice. NRC IN 87-67 recommended plant-specific condition monitoring of masonry walls and administrative controls to ensure that the evaluation basis developed in response to NRC IEB 80-11 is not invalidated by (a) deterioration of the masonry walls (e.g., new cracks not considered in the reevaluation), (b) physical plant changes such as installation of new safety-related systems or components in close proximity to masonry walls, or (c) reclassification of systems or components from non-safety-related to safety-related, provided appropriate evaluation is performed to account for such occurrences.

Important elements in the evaluation of many masonry walls during the NRC IEB 80-11 program included (a) installation of steel edge supports to provide a sound technical basis for boundary conditions used in seismic analysis and (b) installation of steel bracing to ensure stability or containment of unreinforced masonry walls during a seismic event. Consequently, in addition to the development of cracks in the masonry walls, loss of function of the structural steel supports and bracing would also invalidate the evaluation basis. The steel edge supports and steel bracings are considered component supports and aging effects are managed by the Structures Monitoring program (XI.S6).

The program requires periodic visual inspection of masonry walls in the scope of license renewal to detect loss of material and cracking of masonry units and mortar. The aging effects that could impact masonry wall intended function or potentially invalidate its evaluation basis are entered in the corrective action process for further analysis, repair, or replacement.

Since the issuance of NRC IEB 80-11 and NRC IN 87-67, the NRC promulgated 10 CFR 50.65, the Maintenance Rule. Masonry walls may be inspected as part of the "Structures Monitoring Program" (XI.S6) conducted for the Maintenance Rule, provided the 10 attributes described below are incorporated in AMP XI.S6. The aging effects on masonry walls that are considered fire barriers also are managed by XI.M26, Fire Protection.

**Evaluation and Technical Basis**

**1. *Scope of Program:*** The scope includes all masonry walls identified as performing intended functions in accordance with 10 CFR 54.4. The aging effects on masonry walls that are considered fire barriers also are managed by XI.M26, Fire Protection, as well as being managed by this program.

**2. *Preventive Action:*** This is a condition monitoring program and no specific preventive actions are required.

**3. *Parameters Monitored or Inspected:*** The primary parameters monitored are potential shrinkage and/or separation and cracking of masonry walls and gaps between the supports and masonry walls that could impact the intended function or potentially invalidate its evaluation basis.

**4. *Detection of Aging Effects:*** Visual examination of the masonry walls by qualified inspection personnel is sufficient. The frequency of inspection is ~~every 5 years~~ selected with provisions for more frequent inspections in areas where significant loss of material or cracking is observed to ensure there is no loss of intended function between inspections. However, masonry walls that are fire barriers are visually inspected in accordance with XI.M26.

**5. *Monitoring and Trending:*** Trending is not required. Condition monitoring for evidence of shrinkage and/or separation and cracking is achieved by periodic examination. Degradation detected from monitoring is evaluated.

**6. *Acceptance Criteria:*** For each masonry wall, the extent of observed shrinkage and/or separation and cracking of masonry may not invalidate the evaluation basis or impact the wall's intended function. However, further evaluation is conducted if the extent of cracking and loss of material is sufficient to impact the intended function of the wall or invalidate its evaluation basis.

**7. *Corrective Actions:*** A corrective action option is to develop a new analysis or evaluation basis that accounts for the degraded condition of the wall (i.e., acceptance by further evaluation). Other alternatives include repair or replacing the degraded wall. 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:*** Since 1980, masonry walls that perform an intended function have been systematically identified through licensee programs in response to NRC IEB 80-11, NRC Generic Letter 87-02, and 10 CFR 50.48. NRC IN 87-67 documented lessons learned from the NRC IEB 80-11 program and provided recommendations for administrative controls and periodic inspection to ensure that the evaluation basis for each safety-significant masonry wall is maintained. NUREG-1522 documents instances of observed cracks and other deterioration of masonry-wall joints at nuclear power plants. Whether conducted as a stand-alone program or as part of structures monitoring for management review, a masonry wall AMP that incorporates the recommendations delineated in NRC IN 87-67 should ensure that the intended functions of all masonry walls within the scope of license renewal are maintained for the period of extended operation.

## References

10 CFR Part 50, Appendix B, *Quality Assurance Criteria for Nuclear Power Plants*, Office of the Federal Register, National Archives and Records Administration, 2009.

10 CFR 50.48, *Fire Protection,* Office of the Federal Register, National Archives and Records Administration, 2009.

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, 2009.

10 CFR 54.4, *Scope*, Office of the Federal Register, National Archives and Records Administration, 2009.

NRC Generic Letter 87-02, *Verification of Seismic Adequacy of Mechanical and Electrical Equipment in Operating Reactors, Unresolved Safety Issue (USI) A-46,* U.S. Nuclear Regulatory Commission, February 19, 1987.

NRC IE Bulletin 80-11, *Masonry Wall Design,* U.S. Nuclear Regulatory Commission, May 8, 1980.

NRC Information Notice 87-67, *Lessons Learned from Regional Inspections of Licensee Actions in Response to IE Bulletin 80-11*, U.S. Nuclear Regulatory Commission, December 31, 1987.

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.

NUREG-1522, *Assessment of Inservice Condition of Safety-Related Nuclear Power Plant Structures,* June 1995.

## XI.S6 STRUCTURES MONITORING

XI.S6 – 1 Program Description and Elements 3 and 4: Cracking of high strength bolts is not supported by the OE cited for structural bolts. Existing OE is for certain material, size, torque and lubricant applications only and should not be generically applied to all high strength bolts of 150 ksi or more. The Operating Experience cited in NUREG 1801 stated "SCC has occurred in high strength bolts used for nuclear steam supply system component supports (EPRI NP-5769)". The above is cited as operating experience in XI.M18, XI.S1, XI.S3, XI.S6, and XI.S7. The OE cited in NUREG 1801 is NP-5769 (issued in 1988) and was found only in certain specific materials and specifically for NSSS component supports and should not be generically applied to all structural high strength bolts. In certain cases the failures noted were attributed contributing factors including use of molybdenum disulfide thread lubricant which is considered a corrosive environment and is not used in structural steel installations. While EPRI NP-5769 Volume 1, Table 11-1 does list A490 bolts for ductile failures and failure due improper torque, and one instance of a special 4140 material with 200 ksi minimum yield where the A490 specification was used for heat treatment requirements (not an A490 structural bolt) where SCC was noted and associated with a high preload and borated water environment. No SCC failures were noted for commonly used materials in Structural Steel bolting including A307, A325 or A490 bolts in a structural steel application.

Industry documents including NUREG-1339 have not identified any determination specifically as to the ASTM A-490 material's susceptibility to SCC, but rather a determination of the yield stress level at which generic materials should be considered for the possibility of SCC vulnerability. In order for SCC to occur in high strength bolting, three parameters must exist; (1) a corrosive environment, (2) a susceptible material, and (3) high sustained tensile stresses. The absence of any one of these three parameters eliminates the material's susceptibility to SCC. High Strength Structural Bolts including A490 bolts are not subject to high-sustained preload stress and lubricants containing contaminants, such as $MoS_2$, are not approved for use. Therefore, stress corrosion cracking is not considered an applicable aging effect requiring management. The structural bolting wil continue to be visually inspected for loss of preload due to self loosening and loss of material due to corrosion and also visually monitor the associated structural steel and connections for loss of material or other adverse conditions.

The high strength bolts used for structural applications are A325 or A490 Bolts. A325 bolts have a minimum yield of 92 ksi (unlikely to have an actual yield of 150 ksi or more), and A490 bolts have a minimum yield of 130 ksi (potential that some A490 may reach actual yield of 150 ksi or more). Most structural bolts are 1" or less in diameter, however bolts in large girders or supporting large equipment may be over 1" diameter. Test reports are not generally traceable to installation locations.

The AISC "Guide to Design for Bolted and Riveted Joints" Second Edition published in 2001 addresses this concern in section 4.8 and concludes that the tests indicate that black (not galvanized) A490 bolts can be used without problems from "brittle" failures (failures due to hydrogen stress cracking or stress corrosion cracking) in most environments. It was concluded that galvanized A490 bolts should not be used in structures. This same section also concluded that black and galvanized A325 bolts behave satisfactorily with regard to hydrogen stress and stress corrosion cracking in most corrosive environments. AISC publications do not recommend or require volumetric or surface examinations of installed structural bolts for stress corrosion cracking.

We recommend that this new requirement be removed or at the most limited to the specific bolting material types identified in the OE cited on NSSS supports. Volumetric or surface examinations for cracking of high strength bolts should not be generically imposed for all bolts with actual yield strength of 150 ksi or more and greater than 1" diameter. Structural high strength bolts, including A325 and A490 bolts do not have a history of stress corrosion cracking in most environments.

XI.S6 – 2 Element 2: This program is a condition monitoring program and should not provide preventive actions.

XI.S6 – 3 Element 4:

a) Industry guidance found in NUREG-1522, NEI 96-03, and ACI 349 recommend a 5 to 10 year frequency. Program inspection frequency should be addressed as a part of Maintenance Rule implementation. Industry guidance NUREG-1522, "Assessment of Inservice Conditions of Safety-Related Nuclear Plant Structures, and NEI 96-03, "INDUSTRY GUIDELINE FOR MONITORING THE CONDITION OF STRUCTURES AT NUCLEAR POWER PLANTS" recommends a 5 to 10 year frequency. ACI 349 provides guidance for inspecting some structures at greater than a 5 year frequency.

Plants have established their maintenance rule program including inspection frequency based on Maintenance Rule in accordance with 10CFR 50.65, "Requirements for monitoring the effectiveness of maintenance at nuclear power plants". 10CFR 50.65, "MR", does not require 5 year inspection frequency for all the structures. The frequencies of the inspections are established based on risk-informed evaluation process relative to their significant to public health and safety. As such, some structures have inspection frequency between 5 to 10 years. And, some structures based in their site specific OE are inspected even more frequent than every 5 years. Some structures are already in 5 year inspection frequency. For example, Inspection of Water Control Structures governed by RG 1.127 are already conducted on a 5 year or more frequently if conditions warrant as required by RG 1.127. Concrete Containment IWL inspections are also generally performed at a 5 year frequency.

We recommend that this new requirement be removed. Imposing 5 year inspection frequency to all structures and masonry walls seems to be extreme measure without any bases. The inspection frequency should be established based on service and condition assessment of each structure in accordance with Maintenance Rule 10CFR 50.65. That could vary from 5 to 10 years.

b) The ACI 349.3R provides more restrictive qualifications requirements of personnel than the current commitments of most plants under their current licenses. Qualifications that are similar to the guidelines of ACI 349.3 R should also be acceptable as they have been under the current license basis. We recommend that this new requirement be removed.

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## XI.S6 STRUCTURES MONITORING

### 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. Many license renewal applicants have found it necessary to enhance their structures monitoring program to ensure that the aging effects of structures and components in the scope of 10 CFR Part 54.4 are adequately managed during the period of extended operation.

The structures monitoring program consists of periodic visual inspections by personnel qualified to monitor structures and components for applicable aging effects, such as those described in the American Concrete Institute Standards (ACI) 349.3R, ACI 201.1R, and Structural Engineering Institute/American Society of Civil Engineers Standard (SEI/ASCE) 11. ~~Visual inspections should be supplemented with volumetric or surface examinations to detect stress corrosion cracking (SCC) in high strength (actual measured yield strength greater than or equal to 150 kilo-pound per square inch [ksi] or greater than or equal to 1,034 MPa) structural bolts greater than 1 inch (25 mm) in diameter.~~ Identified aging effects are evaluated by qualified personnel using criteria derived from industry codes and standards contained in the plant current licensing bases, including ACI 349.3R, ACI 318, SEI/ASCE 11, and the American Institute of Steel Construction (AISC) specifications, as applicable.

~~The program includes preventive actions delineated in NUREG-1339 and in Electric Power Research Institute (EPRI) NP-5769, NP-5067, and TR-104213 to ensure structural bolting integrity.~~ The program also includes periodic sampling and testing of ground water and the need to assess the impact of any changes in its chemistry on below grade concrete structures.

### Evaluation and Technical Basis

**1.** ***Scope of Program:*** The scope of the program includes all structures, structural components, component supports, and structural commodities in the scope of license renewal that are not covered by other structural AMPs (i.e., "ASME Section XI, Subsection IWE" (XI.S1); "ASME Section XI, Subsection IWL" (XI.S2); "ASME Section XI, Subsection IWF" (XI.S3); "Masonry Walls" (XI.S5); and NRC RG 1.127, "Inspection of Water-Control Structures Associated with Nuclear Power Plants" (XI.S7). Examples of structures, structural components, and commodities in the scope of the program are concrete and steel structures, structural bolting, anchor bolts and embedments, component support members, pipe whip restraints and jet impingement shields, transmission towers, panels and other enclosures, racks, sliding surfaces, sump and pool liners, electrical cable trays and conduits, trash racks associated with water control structures, electrical duct banks, manholes, doors, penetration seals, and tube tracks. The applicant is to specify other structures or components that are in the scope of its structures monitoring program. The scope of this program includes periodic sampling and testing of ground water and may include inspection of masonry walls and water-control structures provided all the attributes of "Masonry Walls" (XI.S5) and NRC RG 1.127, "Inspection of Water-Control Structures Associated with Nuclear Power Plants" (XI.S7) are incorporated in the attributes of this program.

**2. *Preventive Action:*** <u>No preventive actions are specified.</u>~~The structures monitoring program is a condition monitoring program. The program should include preventive actions delineated in NUREG-1339 and in EPRI NP-5769, NP-5067, and TR-104213 to ensure structural bolting integrity. These actions emphasize proper selection of bolting material, lubricants, and installation torque or tension to prevent or minimize loss of bolting preload and cracking of high strength bolting.~~

**3. *Parameters Monitored or Inspected:*** For each structure/aging effect combination, the specific parameters monitored or inspected depend on the particular structure, structural component, or commodity. Parameters monitored or inspected are commensurate with industry codes, standards, and guidelines and also consider industry and plant-specific operating experience. ACI 349.3R and ANSI/ASCE 11 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).

For concrete structures, parameters monitored include loss of material, cracking, increase in porosity and permeability, loss of foundation strength, and reduction in concrete anchor capacity due to local concrete degradation. Steel structures and components are monitored for loss of material due to corrosion. Structural bolting is monitored for loose bolts, missing or loose nuts, and other conditions indicative of loss of preload. ~~High strength (actual measured yield strength ≥ 150 ksi or 1,034 MPa) structural bolts greater than 1 inch (25 mm) in diameter are monitored for SCC.~~ Anchor bolts are monitored for loss of material, loose or missing nuts, and cracking of concrete around the anchor bolts. Accessible sliding surfaces are monitored for indication of significant loss of material due to wear or corrosion, debris, or dirt. Elastomeric vibration isolators and structural sealants are monitored for cracking, loss of material, and hardening. These parameters and other monitored parameters are selected to ensure that aging degradation leading to loss of intended functions will be detected and the extent of degradation can be determined. Ground water chemistry (pH, chlorides, and sulfates) are monitored periodically to assess its impact, if any, on below grade concrete structures. If necessary for managing settlement and erosion of porous concrete sub-foundations, the continued functionality of a site de-watering system is monitored. The plant-specific structures monitoring program should contain sufficient detail on parameters monitored or inspected to conclude that this program attribute is satisfied.

**4. *Detection of Aging Effects:*** Structures are monitored under this program using periodic visual inspection of each structure/aging effect combination by a qualified inspector to ensure that aging degradation will be detected and quantified before there is loss of intended functions. ~~Visual inspection of high strength structural bolting greater than 1 inch (25 mm) in diameter is supplemented with volumetric or surface examinations to detect cracking.~~ Visual inspection of elastomeric vibration isolation elements should be supplemented by feel to detect hardening if the vibration isolation function is suspect. The inspection frequency depends on safety significance and the condition of the structure as specified in NRC RG 1.160, Rev. 2. However, ~~all structures and~~ ground water ~~are~~ <u>is</u> monitored on a frequency not to exceed 5 years. The

program includes provisions for more frequent inspections of structures and components categorized as (a)(1) in accordance with 10 CFR 50.65. Inspector qualifications should be consistent with industry guidelines and standards and guidelines for implementing the requirements of 10 CFR 50.65. Although not required, Q qualifications of inspection and evaluation personnel specified in ACI 349.3R are acceptable for license renewal.
The structures monitoring program addresses detection of aging affects for inaccessible, below-grade concrete structural elements. For plants with non-aggressive ground water/soil (pH > 5.5, chlorides < 500 ppm, or sulfates <1500 ppm), the program recommends: (a) evaluating 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 and (b) examination of representative samples of the exposed portions of the below grade concrete, when excavated for any reason.

For plants with aggressive groundwater/soil (pH < 5.5, chlorides > 500 ppm, or sulfates > 1500 ppm) and/or where the concrete structural elements have experienced degradation, a plant-specific AMP accounting for the extent of the degradation experienced should be implemented to manage the concrete aging during the period of extended operation.

**5. *Monitoring and Trending:*** Regulatory Position 1.5, "Monitoring of Structures," in NRC 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:*** The structures monitoring program calls for inspection results to be evaluated by qualified engineering personnel based on acceptance criteria selected for each structure/aging effect to ensure that the need for corrective actions are indentified before loss of intended functions. The criteria are derived from design bases codes and standards that include ACI 349, ACI 318, SEI/ASCE 11, or the AISC specifications, as applicable, and consider industry and plant operating experience. The criteria are directed at the identification and evaluation of degradation that may affect the ability of the structure or component to perform its intended function. Applicants who are not committed to ACI 349 and elect to use plant-specific criteria for concrete structures should describe the criteria and provide a technical basis for deviations from ACI 349. Loose bolts and nuts and cracked high strength bolts are not acceptable unless accepted by engineering evaluation. Structural sealants are acceptable if the observed loss of material, cracking, and hardening will not result is loss of sealing. Elastomeric vibration isolation elements are acceptable if there is no loss of material, cracking, or hardening that could lead to the reduction or loss of isolation function. Acceptance criteria for sliding surfaces are (a) no indications of excessive loss of material due to corrosion or wear and (b) no debris or dirt that could restrict or prevent sliding of the surfaces as required by design. The structures monitoring program is to contain sufficient detail on acceptance criteria to conclude that this program attribute is satisfied.

**7. *Corrective Actions:*** Evaluations are performed for any inspection results that do not satisfy established criteria. Corrective actions are initiated in accordance with the corrective action process if the evaluation results indicate there is a need for a repair or replacement. 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:*** Although in many plants structures monitoring programs have only recently been implemented, plant maintenance has been ongoing since initial plant operations. NUREG-1522 documents the results of a survey sponsored in 1992 by the Office of Nuclear Regulatory Research to obtain information on the types of distress in the concrete structures, the type of repairs performed, and the durability of the repairs. Licensees who responded to the survey reported cracking, scaling, and leaching of concrete structures. The degradation was attributed to drying shrinkage, freeze-thaw, and abrasion. The NUREG also describes the results of NRC staff inspections at six plants. The staff observed concrete degradation, corrosion of component support members and anchor bolts, cracks and other deterioration of masonry walls, and ground water leakage and seepage into underground structures. The observed and reported degradations were more severe at coastal plants than those observed in inland plants as a result of brackish and sea water. Previous license renewal applicants reported similar degradation and corrective actions taken through their structures monitoring program. There is reasonable assurance that implementation of the structures monitoring program described above will be effective in managing the aging of the in-scope structures and component supports through the period of extended operation.

**References**

10 CFR Part 50, Appendix B, *Quality Assurance Criteria for Nuclear Power Plants*, Office of the Federal Register, National Archives and Records Administration, 2009.

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, 2009.

10 CFR 54.4, *Scope*, Office of the Federal Register, National Archives and Records Administration, 2009.

ACI Standard 201.1R, *Guide for Making a Condition Survey of Concrete in Service,* American Concrete Institute, 1992.

ACI Standard 318, *Building Code Requirements for Reinforced Concrete and Commentary,* American Concrete Institute.

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

AISC, *AISC Specification for Steel Buildings,* American Institute of Steel Construction, Inc., Chicago, IL.

ANSI/ASCE 11-90, 99, *Guideline for Structural Condition Assessment of Existing Buildings,* American Society of Civil Engineers.

EPRI NP-5067, *Good Bolting Practices, A Reference Manual for Nuclear Power Plant Maintenance Personnel*, Volume 1: Large Bolt Manual, 1987; Volume 2: Small Bolts and Threaded Fasteners, Electric Power Research Institute, 1990.

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

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

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

NRC Regulatory Guide 1.142, Rev. 2, *Safety-related Concrete Structures for Nuclear Power Plants (Other than Reactor Vessels and Containments)*, U.S. Nuclear Regulatory Commission, November 2001.

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.

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

NUREG-1522, *Assessment of Inservice Condition of Safety-Related Nuclear Power Plant Structures,* June 1995.

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

XI.S7 – 1 References: These references do not apply to this program.

XI.S7 – 2 Element 2: This program is a condition monitoring program and should not provide preventive actions.

XI.S7 – 3 Element 3, 4 & 6: Cracking of high strength bolts is not supported by the OE cited for structural bolts. Existing OE is for certain material, size, torque and lubricant applications only and should not be generically applied to all high strength bolts of 150 ksi or more. The Operating Experience cited in NUREG 1801 stated "SCC has occurred in high strength bolts used for nuclear steam supply system component supports (EPRI NP-5769)". The above is cited as operating experience in XI.M18, XI.S1, XI.S3, XI.S6, and XI.S7. The OE cited in NUREG 1801 is NP-5769 (issued in 1988) and was found only in certain specific materials and specifically for NSSS component supports and should not be generically applied to all structural high strength bolts. In certain cases the failures noted were attributed contributing factors including use of molybdenum disulfide thread lubricant which is considered a corrosive environment and is not used in structural steel installations. While EPRI NP-5769 Volume 1, Table 11-1 does list A490 bolts for ductile failures and failure due improper torque, and one instance of a special 4140 material with 200 ksi minimum yield where the A490 specification was used for heat treatment requirements (not an A490 structural bolt) where SCC was noted and associated with a high preload and borated water environment. No SCC failures were noted for commonly used materials in Structural Steel bolting including A307, A325 or A490 bolts in a structural steel application.

Industry documents including NUREG-1339 have not identified any determination specifically as to the ASTM A-490 material's susceptibility to SCC, but rather a determination of the yield stress level at which generic materials should be considered for the possibility of SCC vulnerability. In order for SCC to occur in high strength bolting, three parameters must exist; (1) a corrosive environment, (2) a susceptible material, and (3) high sustained tensile stresses. The absence of any one of these three parameters eliminates the material's susceptibility to SCC. High Strength Structural Bolts including A490 bolts are not subject to high-sustained preload stress and lubricants containing contaminants, such as $MoS_2$, are not approved for use. Therefore, stress corrosion cracking is not considered an applicable aging effect requiring management. The structural bolting wil continue to be visually inspected for loss of preload due to self loosening and loss of material due to corrosion and also visually monitor the associated structural steel and connections for loss of material or other adverse conditions.

The high strength bolts used for structural applications are A325 or A490 Bolts. A325 bolts have a minimum yield of 92 ksi (unlikely to have an actual yield of 150 ksi or more), and A490 bolts have a minimum yield of 130 ksi (potential that some A490 may reach actual yield of 150 ksi or more). Most structural bolts are 1" or less in diameter, however bolts in large girders or supporting large equipment may be over 1" diameter. Test reports are not generally traceable to installation locations.

The AISC "Guide to Design for Bolted and Riveted Joints" Second Edition published in 2001 addresses this concern in section 4.8 and concludes that the tests indicate that black (not galvanized) A490 bolts can be used without problems from "brittle" failures (failures due to hydrogen stress cracking or stress corrosion cracking) in most environments. It was concluded that galvanized A490 bolts should not be used in structures. This same section also concluded that black and galvanized A325 bolts behave satisfactorily with regard to hydrogen stress and stress corrosion cracking in most corrosive environments. AISC publications do not recommend or require volumetric or surface examinations of installed structural bolts for stress corrosion cracking.

We recommend that this new requirement be removed or at the most limited to the specific bolting material types identified in the OE cited on NSSS supports. Volumetric or surface examinations for cracking of high strength bolts should not be generically imposed for all bolts with actual yield strength of 150 ksi or more and greater than 1" diameter. Structural high strength bolts, including A325 and A490 bolts do not have a history of stress corrosion cracking in most environments.

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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 raw water-control structures associated with emergency cooling water systems or flood protection of nuclear power plants. The NRC 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 NRC 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.

NRC 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. NRC RG 1.127 delineates current NRC practice in evaluating inservice inspection programs for water-control structures.

For plants not committed to NRC RG 1.127, Revision 1, aging management of water-control structures may be included in the "Structures Monitoring Program" (XI.S6). Even if a plant is committed to NRC 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, as described herein, are incorporated in XI.S6 program attributes.

NRC RG 1.127 attributes evaluated below do not include inspection of dams. For dam inspection and maintenance, programs under the regulatory jurisdiction of the Federal Energy Regulatory Commission (FERC) or the U.S. Army Corps of Engineers, continued through the period of extended operation, are 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 evaluates the effectiveness of the aging management program (AMP) based on compatibility to the common practices of the FERC and Corps programs.

**Evaluation and Technical Basis**

**1. *Scope of Program:*** NRC RG 1.127 applies to raw 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. The scope of the program also includes structural steel and structural bolting associated with water-control structures, steel or wood piles and sheeting required for the stability of embankments and channel slopes, and miscellaneous steel, such as sluice gates and trash racks.

**2. *Preventive Action:*** NRC RG 1.127 is a condition monitoring program. <u>No preventive actions are specified.</u> NRC RG 1.127 is a condition monitoring program. ~~This program is augmented to incorporate preventive measures recommended in NUREG-1339, Electric Power Research Institute (EPRI) TR-104213, EPRI NP-5067, and EPRI NP-5769 to ensure structural bolting integrity. The documents provide guidelines for selection of replacement bolting material, approved thread lubricants, and appropriate torque and preload to be used for installation of bolting.~~

**3. *Parameters Monitored or Inspected:*** NRC 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 are those described in American Concrete Institute (ACI) 201.1 and ACI-349-3R. These include cracking, movements (e.g., settlement, heaving, deflection), conditions at junctions with abutments and embankments, loss of material, increase in porosity and permeability, 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.

Steel components are monitored for loss of material due to corrosion.

Parameters monitored for channels and canals include erosion or degradations that may impose constraints on the function of the cooling system and present a potential hazard to the safety of the plant. Submerged emergency canals (e.g., artificially dredged canals at the river bed or the bottom of the reservoir) should be monitored for sedimentation, debris, or instability of slopes that may impair the function of the canals under extreme low flow conditions.

Further details of parameters to be monitored and inspected for these and other water-control structures are specified in Section C.2 of NRC RG 1.127. ~~The program is augmented to require monitoring of bolted connections for loss of material and loose bolts and nuts and other conditions indicative of loss of preload. High strength (actual measured yield strength ≥ 150 ksi or 1,034 MPa) structural bolts greater than 1 inch (25 mm) in diameter are monitored for stress corrosion cracking.~~ The program is ~~also~~ augmented to require monitoring of wooden components for loss of material and change in material properties

**4. *Detection of Aging Effects:*** NRC RG 1.127 specifies that inspection of water-control structures should be conducted under the direction of qualified engineers experienced in the investigation, design, construction, and operation of these types of facilities. Visual inspections are primarily used to detect degradation of water-control structures. ~~Visual inspection of high strength structural bolting greater than 1 inch (25 mm) in diameter should be supplemented with volumetric or surface examinations to detect cracking. This requirement can be waived with adequate technical justification.~~ In some cases, instruments have been installed to measure the behavior of water-control structures. NRC 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. NRC RG 1.127 describes periodic inspections to be performed at least once every 5 years. This interval has been shown to be adequate to detect degradation of water-control structures before a loss of an intended function. The program should include provisions for increased inspection frequency if the extent of the degradation is such that the structure or component may not meet its design basis if allowed to continue uncorrected until the next normally scheduled inspection. NRC 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.

The program should address detection of aging affects for inaccessible, below-grade, and submerged concrete structural elements. For plants with non-aggressive raw water and groundwater/soil (pH > 5.5, chlorides < 500 parts per million [ppm], or sulfates < 1500 ppm), the program should require (a) evaluating 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 and (b) examination of representative samples of the exposed portions of the below-grade concrete when excavated for any reason. Submerged concrete structures should be inspected during periods of low tide or when dewatered and accessible.

For plants with aggressive environment raw water (pH < 5.5, chlorides > 500 ppm, or sulfates > 1500 ppm) or ground water/soil and/or where the concrete structural elements have experienced degradation, a plant-specific AMP accounting for the extent of the degradation experienced should be implemented to manage the concrete aging during the period of extended operation.

**5. *Monitoring and Trending:*** Water-control structures are monitored by periodic inspection as described in NRC RG 1.127. Changes of degraded conditions from prior inspection, such as growth of an active crack or extent of corrosion, should be trended until it is evident the change is no longer occurring or until corrective actions are implemented in accordance with 10 CFR 50.65 and RG 1.160, Rev. 2.

**6. *Acceptance Criteria:*** Quantitative acceptance criteria to evaluate the need for corrective actions are not specified in NRC RG 1.127. However, the "Evaluation Criteria" provided in

Chapter 5 of ACI 349.3R 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 are acceptable. Acceptance criteria for earthen structures such as canals, and embankments are consistent with programs falling within the regulatory jurisdiction of the FERC or the U.S. Army Corps of Engineers. Loose bolts and nuts, ~~cracked high-strength bolts,~~ and degradation of piles and sheeting are accepted by engineering evaluation or subject to corrective actions. Engineering evaluation should be documented and based on codes, specifications, and standards such as AISC specifications, SEI/ASCE 11, and those referenced in the plant's current licensing.

**7. *Corrective Actions:*** NRC RG 1.127 recommends that when inspection 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 NRC RG 1.127 programs, at a number of nuclear power plants, and in some cases, it has required remedial action. NRC NUREG-1522 described instances and corrective actions of severely degraded steel and concrete components at the intake structure and pumphouse of coastal plants. Other degradations described in the NUREG include appreciable leakage from the spillway gates, concrete cracking, corrosion of spillway bridge beam seats of a plant dam and cooling canal, and appreciable differential settlement of the outfall structure of another. 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 NRC RG 1.127 have been successful in detecting significant degradation before loss of intended function occurs.

**References**

10 CFR Part 50, Appendix B, *Quality Assurance Criteria for Nuclear Power Plants*, Office of the Federal Register, National Archives and Records Administration, 2009.

ACI Standard 201.1R, *Guide for Making a Condition Survey of Concrete in Service,* American Concrete Institute, 1992.

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

~~EPRI NP-5067, Good Bolting Practices, A Reference Manual for Nuclear Power Plant Maintenance Personnel, Volume 1: Large Bolt Manual, 1987; Volume 2: Small Bolts and Threaded Fasteners, Electric Power Research Institute, 1990.~~

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

~~EPRI TR-104213, Bolted Joint Maintenance & Application Guide, Electric Power Research Institute, December 1995.~~

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

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

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

NUREG-1522, *Assessment of Inservice Conditions of Safety-Related Nuclear Plant Structures*, U.S. Nuclear Regulatory Commission, June 1995.

## XI.S8 PROTECTIVE COATING MONITORING AND MAINTENANCE PROGRAM

XI.S8 – 1 Program Description and Elements 1, 3, and 10: A coatings program developed to previously approved, and adopted by current licensing basis to in accordance with Regulatory Guide 1.54 Rev. 0 should also meet the aging management program requirements and should be acceptable here as well.

Reg. Guide 1.54, Rev 0 is the current licensing basis at a number of plants. The new requirement, compliance with Reg. Guide 1.54 Rev 1 (and later), requires an expanded coating program, and the Rev 2 document (soon to be issued) will require expanded resources including an ASTM qualified "Coating Specialist" at each site. Rev 2 also requires full compliance to ASTM standards that were recently written to support new power plants. We believe the ASTM Committee D33 had no intention that these would be a back fit applied to existing licensed power plants. New ASTM standards have not been reviewed for gaps against ANSI requirements. The current licensing basis commitments are considered appropriate to detect and address coatings condition.

We recommend revising this new requirement to state a coating program developed to the previously approved Regulatory Guide 1.54 Rev. 0 and improved in response to Generic Letter 98-04 and implemented per the Maintenance Rule meets the current licensing basis and should be acceptable in GALL Rev. 2. Otherwise, for license renewal this new requirement will force the licensee to take exception to GALL.

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## XI.S8 PROTECTIVE COATING MONITORING AND MAINTENANCE PROGRAM

### 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~~ 0 , or latest version) 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 Emergency Core Cooling Systems (ECCS) suction strainers, which reduces flow through the system and could cause unacceptable head loss for the pumps.

Maintenance of Service Level I coatings applied to carbon steel and concrete surfaces inside containment (e.g., steel liner, steel containment shell, structural steel, supports, penetrations, and concrete walls and floors) also serve to prevent or minimize loss of material due to corrosion of carbon steel components and aids in decontamination. Regulatory Position C4 in NRC RG 1.54, Rev. ~~1~~ 0, 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. American Society for Testing of Materials (ASTM) D 5163-08 and endorsed years of the standard in NRC RG 1.54 are acceptable and considered consistent with NUREG-1801. In addition, Electric Power Research Institute (EPRI) Report

1003102, Guidelines for Inspection and Maintenance of Safety-related Protective Coatings, provides additional information on the ASTM standard guidelines.

A comparable program for monitoring and maintaining protective coatings inside containment, developed in accordance with NRC RG 1.54, Rev.~~1~~ 0, is acceptable as an aging management program for license renewal.

Service Level I coatings credited for preventing corrosion of steel containments and steel liners for concrete containments are subject to requirements specified by the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel (B&PV) Code, Section XI, Subsection IWE (XI.S1). However, this program (XI.S8) reviews Service Level I coatings to ensure that the protective coating monitoring and maintenance program are adequate for license renewal.

**Evaluation and Technical Basis**

**1.** ***Scope of Program:*** The minimum scope of the program is Service Level I coatings applied to steel and concrete surfaces inside containment (e.g., steel liner, steel containment shell, structural steel, supports, penetrations, and concrete walls and floors), defined in NRC RG 1.54, Rev ~~1~~ 0, 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." The scope of the program also should include any Service Level I coatings that are credited by the licensee for preventing loss of material due to corrosion in accordance with XI.S1.

**2.** ***Preventive Action:*** The program is a condition monitoring program and does not recommend any preventive actions. However, for plants that credit coatings to minimize loss of material, this program is a preventive action.

**3.** ***Parameters Monitored or Inspected:*** Regulatory Position C4 in NRC RG 1.54, Rev ~~1~~ 0, states that "ASTM D 5163-96 provides guidelines that are acceptable to the NRC staff for establishing an in-service coatings monitoring program for Service Level I coating systems in operating nuclear power plants..." ASTM D 5163-96 has been superseded by ASTM D 5163-08. ASTM D 5163-08 identifies the parameters monitored or inspected to be "any visible defects, such as blistering, cracking, flaking, peeling, rusting, and physical damage."

**4.** ***Detection of Aging Effects:*** ASTM D 5163-08, paragraph 6, defines the inspection frequency to be each refueling outage or during other major maintenance outages, as needed. ASTM D 5163-08, paragraph 9, discusses the qualifications for inspection personnel, the inspection coordinator, and the inspection results evaluator. ASTM D 5163-08, subparagraph 10.1, discusses development of the inspection plan and the inspection methods to be used. It states, "A general visual inspection shall be conducted on all readily accessible coated surfaces during a walk-through. After a walk-through, or during the general visual inspection, thorough visual inspections shall be carried out on previously designated areas and on areas noted as deficient during the walk-through. A thorough visual inspection shall also be carried out on all coatings near sumps or screens associated with the Emergency Core Cooling System (ECCS)." This subparagraph also addresses field documentation of inspection results. ASTM D 5163-08, subparagraph 10.5, identifies instruments and equipment needed for inspection.

**5.** ***Monitoring and Trending:*** ASTM D 5163-08 identifies monitoring and trending activities in subparagraph 7.2, which specifies a pre-inspection review of the previous two monitoring reports, and in subparagraph 11.1.2, which specifies that the inspection report should prioritize repair areas as either needing repair during the same outage or postponed to future outages, but under surveillance in the interim period.

**6. *Acceptance Criteria:*** ASTM D 5163-08, subparagraphs 10.2.1 through 10.2.6, 10.3, and 10.4, contain one acceptable method for characterization, documentation, and testing of defective or deficient coating surfaces. Additional ASTM and other recognized test methods are available for use in characterizing the severity of observed defects and deficiencies. The evaluation covers blistering, cracking, flaking, peeling, delamination, and rusting. ASTM D 5163-08, paragraph 11, addresses evaluation. It specifies that the inspection report is to be evaluated by the responsible evaluation personnel, who prepare a summary of findings and recommendations for future surveillance or repair, including an analysis of reasons or suspected reasons for failure. Repair work is prioritized as major or minor defective areas.

**7. *Corrective Actions:*** A recommended corrective action plan is required for major defective areas so that these areas can be repaired during the same outage, if appropriate. 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*:** NRC Information Notice 88-82, NRC Bulletin 96-03, NRC Generic Letter (GL) 04-02, and NRC GL 98-04 describe industry experience pertaining to coatings degradation inside containment and the consequential clogging of sump strainers. NRC RG 1.54, Rev. 1, was issued in July 2000. However, ~~M~~monitoring and maintenance of Service Level I coatings conducted in accordance with NRC RG 1.54, Rev. 0, Regulatory Position C4 is expected to be an effective program for managing degradation of Service Level I coatings~~.~~, and consequently an effective means to manage loss of material due to corrosion of carbon steel structural elements inside containment.

**References**

10 CFR Part 50, Appendix B, *Quality Assurance Criteria for Nuclear Power Plants*, Office of the Federal Register, National Archives and Records Administration, 2009.

ASTM D 5163-05, *Guide for Establishing Procedures to Monitor the Performance of Coating Service Level I Coating Systems in an Operating Nuclear Power Plant,* American Society for Testing and Materials, 2005.

ASTM D 5163-08, *Standard Guide for Establishing a Program for Condition Assessment of Coating Service Level I Coating Systems in Nuclear Power Plants,* American Society for Testing and Materials, 2008.

ASTM D 5163-96, *Standard Guide for Establishing Procedures to Monitor the Performance of Safety Related Coatings in an Operating Nuclear Power Plant,* American Society for Testing and Materials, 1996.

EPRI Report 1003102, *Guideline on Nuclear Safety-Related Coatings*, Revision 1, (Formerly TR-109937), Electric Power Research Institute, November 2001.

NRC Bulletin 96-03, *Potential Plugging of Emergency Core Cooling Suction Strainers by Debris in Boiling-Water Reactors*, U.S. Nuclear Regulatory Commission, May 6, 1996.

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

NRC Generic Letter 04-02, *Potential Impact of Debris Blockage on Emergency Recirculation during Design Basis Accidents at Pressurized-Water Reactors*, U.S. Nuclear Regulatory Commission, September 13, 2004.

NRC Information Notice 88-82, *Torus Shells with Corrosion and Degraded Coatings in BWR Containments*, U.S. Nuclear Regulatory Commission, November 14, 1988.

NRC Information Notice 97-13, *Deficient Conditions Associated With Protective Coatings at Nuclear Power Plants*, U.S. Nuclear Regulatory Commission, March 24, 1997.

NRC Regulatory Guide 1.54, Rev. 0, *Quality Assurance Requirements for Protective Coatings Applied to Water-Cooled Nuclear Power Plants,* U.S. Nuclear Regulatory Commission, June 1973.

NRC Regulatory Guide 1.54, Rev. 1, Service Level I, II, and III Protective Coatings Applied to Nuclear Power Plants, U.S. Nuclear Regulatory Commission, July 2000.

| NUREG 1801 (GALL) & NUREG 1800 (SRP), Rev. 2, Industry AMR Comments | | | |
|---|---|---|---|
| **#** | **(Page #),Section #** | **Recommended changes (Deletions - ~~Strikethrough~~, Additions – Underline)** | **Justifications** |
| 1 | (Page II A1-5) II.A1.CP-101 | Chapter XI.S2, "ASME Section XI, Subsection IWL", or Chapter XI.S6, "Structure Monitoring" | Evidence of degradation due to this aging effect can also be identified under Structure Monitoring. |
| 2 | (Page II A2-4) II.A2.CP-71, (Page II B3-4) II.B3.1.CP-71 | Chapter XI.S6, "Structures Monitoring", or Chapter XI.S2, "ASME Section XI, Subsection IWL" | Evidence of degradation due to this aging effect may be identified under IWL. |
| 3 | (Page II A2-6) II.A2.CP-69, (Page II B1-6) II.B1.2.CP-105, (Page II B2-7) II.B2.2.CP-105, (Page II B3-7) II.B3.1.CP-69, (Page II B3-12) II.B3.2.CP-105 | Chapter XI.S2, "ASME Section XI, Subsection IWL", or Chapter XI.S6, "Structure Monitoring" | Evidence of degradation due to this aging effect can also be identified under Structure Monitoring. |
| 4 | (Page II A1-7) II.A1.CP-98, (Page II A2-7) II.A2.CP-98, (Page II B1-8) II.B1.2.CP-63, (Page II B2-2) II.B2.1.CP-63, (Page II B2-9) II.B2.2.CP-63 | 4. Borated water spills and water ponding on the concrete floor are not common and when detected are cleaned up or diverted to a sump in a timely manner. | 1) Spills are not common.<br>2) To be consistent with II.B3.2.CP-98. |
| 5 | (Page II A3-3) II.A3.CP-39, (Page II B4-3) II.B4.CP-39 | ~~Chapter XI.S1, "ASME Section XI, Subsection IWE" and,~~ Chapter XI.S4, "10 CFR Part 50, Appendix J" | IWE will not detect loss of leak tightness. |

| NUREG 1801 (GALL) & NUREG 1800 (SRP), Rev. 2, Industry AMR Comments | | | |
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| **#** | **(Page #),Section #** | **Recommended changes (Deletions - ~~Strikethrough~~, Additions – <u>Underline</u>)** | **Justifications** |
| 6 | (Page II A3-3) II.A3.CP-152, (Page II B4-3) II.B4.CP-152, (Page III A4-4) III.A4.TP-301 | Chapter XI.S8, "Protective Coating Monitoring and Maintenance" <u>or plant specific program in response to GL 98 - 04 for those plants not crediting coatings for loss of material</u>. | All plants have developed plant specific program in response to GL 98 - 04 to monitor and maintain condition of containment coatings. |
| 7 | (Page II A3-3) II.A3.CP-150, (Page II B4-3 ) II.B4.CP-150 | ~~Structural~~ Pressure - retaining bolting | Chapter XI.S1, "ASME Section XI, Subsection IWE" applies to containment pressure - retaining bolting only. |
| 8 | (Page II A3-4 ) II.A3.CP-148 | ~~Structural~~ Pressure - retaining bolting | Chapter XI.S1, "ASME Section XI, Subsection IWE" applies to containment pressure - retaining bolting only. |
| 10 | (Page II A3-2) II.A3.CP-36, (Page II B4-2) II.B4.CP-36 | Chapter XI.S1, "ASME Section XI, Subsection IWE" and, Chapter XI.S4, "10 CFR Part 50, Appendix J" (Note: IWE examination category E-F, surface examination of dissimilar metal welds, specified in 1992 edition of ASME Code is recommended) | Examination category E-F does no longer exist on latest edition of the ASME Code. If augmentation to code requirements are necessary they should be addressed in the program of Chapter XI.S1, "ASME Section XI, Subsection IWE" . |
| 11 | (Page III A1-5) III.A1.TP-114, (Page III A2-6) III.A2.TP-114, (Page III A3-6) III.A3.TP-114, (Page III A4-3) III.A4.TP-114, (Page III A5-6) III.A5.TP-114 | Plant-specific aging management program. ~~The implementation of 10 CFR 50.55a and ASME Section XI, Subsection IWL would not be able to identify the reduction of strength and modulus of elasticity due to elevated temperature. Thus, for any portions of concrete containment that exceed specified temperature limits, further evaluations are warranted.~~ Subsection CC-3400 of ASME Section III, Division 2, specifies the concrete temperature limits for normal operation or any other long-term period. The temperatures shall not exceed 150°F except for local areas, such as around penetrations, which are not allowed to exceed 200°F. If significant equipment loads are supported by concrete at temperatures exceeding 150°F, an evaluation of the ability to withstand the postulated design loads is to be made. Higher temperatures than given above may be allowed in the concrete if tests and/or calculations are provided to evaluate the reduction in strength and modulus of elasticity and these reductions are applied to the design calculations. | IWL does not apply to Group I Structures. |

| NUREG 1801 (GALL) & NUREG 1800 (SRP), Rev. 2, Industry AMR Comments | | | |
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| **#** | **(Page #),Section #** | **Recommended changes (Deletions - ~~Strikethrough~~, Additions – Underline)** | **Justifications** |
| 12 | (Page III A1-6) III.A1.TP-300, (Page III A2-7) III.A2.TP-300, (Page III A3-7) III.A3.TP-300, (Page III A4-4) III.A4.TP-300, (Page III A5-7) III.A5.TP-300, (Page III A7-6) III.A7.TP-300 (Page III A8-5) III.A8.TP-300, (Page III A9-6) III.A9.TP-300, (Page III B2-2) III.B2.TP-300 (Page III B3-2) III.B3.TP-300, (Page III B4-2) III.B4.TP-300, (Page III B5-2) III.B5.TP-300 | Delete these line items. | These line items are not supported by OE and should be removed or limited to the specific type of bolting material and sizes where cracking has been found on NSSS supports. It is not warranted to generically extend the limited material specific OE (which may be partially caused by the type of lubricant) to all bolts of 150 ksi and over regardless of material and lubricant. |

| NUREG 1801 (GALL) & NUREG 1800 (SRP), Rev. 2, Industry AMR Comments | | | |
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| **#** | **(Page #),Section #** | **Recommended changes (Deletions - ~~Strikethrough~~, Additions – Underline)** | **Justifications** |
| 13 | (Page III A1-7) III.A1.TP-287, (Page III A2-7) III.A2.TP-287, (Page III A3-7) III.A3.TP-287, (Page III A4-4) III.A4.TP-287, (Page III A5-7) III.A5.TP-287, (Page III A7-6) III.A7.TP-287, (Page III A8-6) III.A8.TP-287, (Page III A9-6) III.A9.TP-287, (Page III B2-2) III.B2.TP-287, (Page III B3-2) III.B3.TP-287, (Page III B4-2) III.B4.TP-287, (Page III B5-2) III.B5.TP-287 | Delete these line items. | These line items are covered under III.A1.TP-248 and III.A1.TP-274. |
| 14 | (Page III A5-7) III.A5.TP-34 | Delete this line item. | The aging effect for the block walls are adequately covered under line item III.A5.T-12. |
| 15 | (Page III A5-3) III.A6.TP-223 | Chapter XI.S6, "Structure Monitoring Program" or Chapter XI.S7, "Regulatory Guide 1.127, Inspection of Water-Control Structures Associated with Nuclear Power Plants" or the FERC / US Army Corp of Engineers dam inspections and maintenance programs. | To maintain option of evaluating wood components under Structures Monitoring. |

| NUREG 1801 (GALL) & NUREG 1800 (SRP), Rev. 2, Industry AMR Comments | | | |
|---|---|---|---|
| **#** | **(Page #),Section #** | **Recommended changes (Deletions - ~~Strikethrough~~, Additions – Underline)** | **Justifications** |
| 16 | (Page III B1-5) III.B1.1.TP-232, (Page III B1-9) III.B1.2.TP-232, (Page III B1-13) III.B1.3.TP-232 | Chapter XI.M2, "Water Chemistry," ~~for BWR water,~~ and Chapter XI.S3, "ASME Section XI, Subsection IWF" | It is self explanatory without referencing to BWR since these line items apply to BWR. |
| 17 | (Page III B2-2) III.B2.TP-41, (Page III B3-2) III.B3.TP-41 | Delete these line items. | This material does not apply to this group (III.B2 and III.B3). |
| 18 | SRP, Table 3.0-1, (Page 3.0-5), ASME Section XI IWE | The ASME Section XI, Subsection IWE program consists of periodic visual, surface, and volumetric inspection of pressure-retaining components of steel and concrete containments for signs of degradation, assessment of damage, and corrective actions. The program also includes aging management for the potential loss of material due to corrosion in the inaccessible areas of the boiling water reactor (BWR) Mark I steel containment~~, and surface examination for the detection of cracking of structural bolting~~. This program is in accordance with ASME Section XI, Subsection IWE, ~~20014~~ <u>2004</u> edition, including the 2002 and 2003 Addenda. | IWE addresses pressure retaining bolting. |
| 19 | SRP Table 3.0-1, (Page 3.0-12), Fire Protection, Chapter XI.M26 | The program includes fire barrier ~~and diesel-driven fire pump~~ inspections. 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 program also includes periodic inspection and test of halon/carbon dioxide fire suppression systems. | To match the scope of XI.M26 Fire Protection Program. |

| NUREG 1801 (GALL) & NUREG 1800 (SRP), Rev. 2, Industry AMR Comments | | | |
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| **#** | **(Page #),Section #** | **Recommended changes (Deletions - ~~Strikethrough~~, Additions – Underline)** | **Justifications** |
| 20 | SRP Table 3.0-1, (Page 3.0-15), Inspection of Overhead Heavy Load and Light Load Handling Related to Refueling) Handling Systems, Chapter XI.M23 | The program evaluates the effectiveness of the maintenance monitoring program and the effects of past and future usage on the structural reliability of cranes and hoists. ~~The number and magnitude of lifts made by the hoist or crane are also reviewed.~~ Rails and girders are visually inspected on a routine basis for degradation; functional tests are performed to assure their integrity. These cranes must also comply with the maintenance rule requirements provided in 10 CFR 50.65. | To match the scope of XI.M23 Inspection of Overhead Heavy Load and Light Load Handling Related to Refueling) Handling Systems. |
| 21 | SRP, Table 3.5-1, (Page 3.5-30) Item 22 | ~~Structural~~ Pressure - retaining bolting, Steel elements: downcomer pipes | Chapter XI.S1, "ASME Section XI, Subsection IWE" applies to containment pressure - retaining bolting only. |
| 22 | SRP, Table 3.5-1, (Page 3.5-40) Item 58 | III.B1.1.TP-41<br>~~III.B2.TP-41~~<br>~~III.B3.TP-41~~ | Delete the items shown crossed out. |
| 23 | SRP, Table 3.5-1, (Page 3.5-42) Item 65 | Delete this line item. | The aging effect for the block walls are adequately covered under line item III.A5.T-12. |
| 24 | SRP, Table 3.5-1, (Page 3.5-47), line Item 77 | 1. ~~High-strength structural bolting,~~ Support members; welds; bolted connections; support anchorage to building structure<br>2. ~~III.A1.TP-287, III.A2.TP-287, III.A3.TP-287, III.A4.TP-287, III.A5.TP-287, III.A7.TP-287, III.A8.TP-287, III.A9.TP-287, III.B2.TP-287, III.B3.TP-287, III.B4.TP-287, and III.B5.TP-287~~ | 1 & 2) These line items are covered under III.A1.TP-248 and III.A1.TP-274. |
| 25 | SRP, Table 3.5-1, (Page 3.5-47) Item 76 | Delete these line items. | These line items are not supported by OE and should be removed or limited to the specific type of bolting material and sizes where cracking has been found on NSSS supports. It is not warranted to generically extend the limited material specific OE (which may be partially caused by the type |

| NUREG 1801 (GALL) & NUREG 1800 (SRP), Rev. 2, Industry AMR Comments | | | |
|---|---|---|---|
| # | (Page #),Section # | Recommended changes (Deletions - ~~Strikethrough~~, Additions – Underline) | Justifications |
| | | | of lubricant) to all bolts of 150 ksi and over regardless of material and lubricant. |

## CHAPTER VI, ELECTRICAL COMPONENTS

VI.A-1 Page VI A-1, System, Structures and Components, Paragraph 2: Remove proposed new third sentence beginning "As specified in..." Revise to remove reference, start at "The electrical distribution..."

VI.A-2 Page VI A-1, System Interfaces, Paragraph 1: Remove proposed change. Make it consistent with Section 3

VI.A-3 Reference paragraphs in VI.A-1. Make changes to account for removing the old SBO language.

## MARK-UP: CHAPTER VI, ELECTRICAL COMPONENTS

VI.A-1 & VI.A-3

*This section also addresses components that are relied upon to meet the station blackout (SBO)* requirements for restoration of offsite power. The offsite power system relied upon in the plant-specific current licensing basis for compliance with 10 CFR 50.63, that is used to connect the plant to the offsite power source, is included in the SBO restoration equipment scope. ~~As specified in U.S. Nuclear Regulatory Commission (NRC) Regulatory Guide (RG) 1.60, "Monitoring the Effectiveness of Maintenance at Nuclear Power Plants,"~~ The electrical distribution equipment out to the first inter-tie with the offsite distribution system (i.e., equipment in the switchyard) should be included within the SBO restoration equipment scope. This path typically includes the ~~switchyard circuit breakers~~ the first inter-tie devices that connect to the offsite system power transformers (startup transformers), the transformers themselves, the intervening overhead or underground circuits between circuit breaker and transformer and transformer and onsite electrical distribution system (including bus ducts or cables), and associated control circuits and structures.

VI.A-2

Electrical cables and connections functionally interface with all plant systems that rely on electric power or instrumentation and control. Electrical cables and connections also interface with and are supported by structural commodities (e.g., cable trays, conduit, cable trenches, cable troughs, duct banks, cable vaults, and manholes) that are reviewed, as appropriate, in the Systems, Structures, and Components section.

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## Section XI.E1, "INSULATION MATERIAL FOR ELECTRICAL CABLES AND CONNECTIONS NOT SUBJECT TO 10 CFR 50.49 ENVIRONMENTAL QUALIFICATION REQUIREMENTS"

XI.E1-1 Page XI E1-1, Program Description, Paragraph 3: Move the 3rd paragraph to the 1st paragraph. Also, spell out the acronym "AMP". Consistency with the format of XI.E2 -XI.E4.

XI.E1-2 Page XI E1-1, Program Description, Paragraph 2: 7th line: revise "or moisture specification over the specified life (if applicable) of the cables or connection..." to read "or moisture conditions for the cables or connection..." Also make connection plural (i.e should read connections). Sentence is now consistent with the last sentence in Element 3.

XI.E1-3 Page XI E1-1, Program Description, Paragraph 2: Last line: revise "plant specific industry operating experience." to read "plant specific and industry operating experience." This was also as submitted in our NEI letter to the NRC concerning this topic

XI.E1-4 Page XI E1-2, Parameters Monitored/Inspected (Element 3), Paragraph 1: line three, change "signs of..." to "indicating we may have..."

XI.E1-5 Page XI E1-2, Parameters Monitored/Inspected (Element 3), Paragraph 1: Delete the word "all". This word would change this effort from reasonable assurance to an absolute assurance that would require more than just a visual inspection.

XI.E1-6 Page XI E1-3, Operating Experience (Element 10), Paragraph 1: line two, the list of environments should read "temperature, radiation, or moisture" [missing commas]

Program Description (XI.E1-1, 2, 3)

| NOTE: The change shown has the old 3rd paragraph moved to new 1st paragraph. The old 1st paragraph or new 2nd paragraph is shown for clarity only. |
|---|

The purpose of the aging management program (AMP) 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 temperature, radiation, or moisture are maintained consistent with the current licensing basis through the period of extended operation.

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.

Insulation materials used in electrical 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 plant design environment for the cable or connection insulation material that could increase the rate of aging of a component or have an adverse effect on operability. An adverse localized environment exists based on the most limiting temperature, radiation, or moisture ~~specification over the specified life (if applicable) of~~ conditions for the cables or connections insulation material. Adverse localized environments can be identified through the use of an integrated approach. This methodology may include, but is not limited to, (a) the review of Environmental Qualification (EQ) zone maps that show radiation levels and temperatures for various plant areas, (b) consultations with plant staff who are cognizant of plant conditions, (c) utilization of infrared thermography to identify hot spots on a real-time basis, and (d) the review of relevant plant specific and industry operating experience.

Parameters Monitored/Inspected (XI.E1-4, 5)
~~All~~ Accessible electrical cables and connections installed in adverse localized environments are visually inspected for cable jacket and connection insulation surface anomalies and ~~signs of~~ indicating we may have *reduced insulation resistance due to thermal/thermoxidative* degradation of organics, radiolysis and photolysis (UV sensitive materials only) of organics; radiation-induced oxidation, and moisture intrusion as indicated by signs of embrittlement,

discoloration, cracking, melting, swelling or surface contamination. An adverse localized environment is a plant-specific condition; therefore, the applicant should clearly define how this condition is determined. The applicant should determine and inspect the adverse localized conditions for each of the most limiting temperature, radiation, or moisture conditions for the accessible cables and connections that are within the scope of license renewal.

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**Section XI.E2, "INSULATION MATERIAL FOR ELECTRICAL CABLES AND CONNECTIONS NOT SUBJECT TO 10 CFR 50.49 ENVIRONMENTAL QUALIFICATION REQUIREMENTS USED IN INSTRUMENTATION CIRCUITS"**

XI.E2-1 Page XI E2-1, Program Description, Paragraph 2: Make the first 2 sentences a stand-alone paragraph. Consistent with XI.E1. Or, conversely, make XI.E1 one long paragraph like XI.E2. The change shown is with the paragraph split.

Program Description (XI.E2-1)

The purpose of this AMP 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 used in instrumentation circuits with sensitive, high voltage, low-level current signals exposed to adverse localized environments caused by temperature, radiation, or moisture) are maintained consistent with the current licensing basis through the period of extended operation.

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 design environment.

Insulation materials used in electrical cables or connections may degrade more rapidly in adverse localized environments. An adverse localized environment is a condition in a limited plant area that is significantly more severe than the plant design environment for the cable or connection insulation material that could increase the rate of aging of a component or have an adverse effect on operability. Exposure of electrical cable and connection insulation material to adverse localized environments caused by temperature, radiation, or moisture can result in reduced insulation resistance (IR). Reduced IR causes an increase in leakage currents between conductors and from individual conductors to ground. A reduction in IR is a concern for all circuits, but especially those with sensitive, high voltage, low-level current signals, such as radiation monitoring and nuclear instrumentation circuits, because a reduced IR may contribute to signal inaccuracies.

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**Section XI.E3, " INACCESSIBLE MEDIUM-VOLTAGE POWER CABLES NOT SUBJECT TO 10 CFR 50.49 ENVIRONMENTAL QUALIFICATION REQUIREMENTS"**

XI.E3-1 Page XI E3-1, Program Description, Paragraph 2: second line, "be" at the end of the line should be "and are"

XI.E3-4 Page XI E3-2, Preventive Actions (Element 2), Paragraph 1: change fourth line "conduit ends and..." to "conduit ends in..."

XI.E3-5 Page XI E3-2, Parameters Monitored/Inspected (Element 3), Paragraph 1: Current change discusses inspection first, then testing. Suggest reversing the order. Cable testing is the primary method for assessing aging of cable insulation

XI.E3-6 Page XI E3-2, Detection of Aging Effects (Element 4), Paragraph 1: First sentence relative to verifying dewatering system operation prior to known or predicted flood events should be deleted. The verification is neither an aging preventive nor an aging detection mechanism. Infrequent submergence (rain and drain) is not a stressor for cable insulation degradation.

XI.E3-7 Page XI E3-2, Detection of Aging Effects (Element 4), Paragraph 1: Second sentence sets maximum frequency at annually, or more frequent based on operating experience. Suggest maximum frequency remain at 2 years.

XI.E3-8 Page XI E3-2, Detection of Aging Effects (Element 4), Paragraph 1: Third sentence, suggest manhole inspections not be required if dewatering equipment is functioning. Sump trouble alarms will provide indications of water accumulation. Sump levels ensure cable is not immersed or submerged. Note that Structures Monitoring Program would not be changed regardless of mechanism for dewatering.

XI.E3-9 Page XI E3-2, Detection of Aging Effects (Element 4), Paragraph 2: First sentence sets test frequency at once every 3 refueling cycles. Suggest test frequency remain at every 10 years, adjusted for test results as determined through the corrective action process. Or at least use every 6 years instead of every 3R.

XI.E3-10 Page XI E3-2, Detection of Aging Effects (Element 4), Paragraph 1: Move to Element 2 and make frequency commensurate with plant operating experience or corrective action program.

XI.E3-13 Page XI E3-3, Operating Experience (Element 10), Paragraph 1: First sentence, remove the word "most"

XI.E3-14 Page XI E3-4, Operating Experience (Element 10), Paragraph 2 & 3: The paragraphs beginning "The NRC..." and "Therefore..." should be deleted. The information is background/historical information about the process rather than operating experience

<u>Program Description (XI.E3-1)</u>
Most electrical cables in nuclear power plants are located in dry environments. However, some cables may be exposed to wetting or submergence, ~~be~~ <u>and are</u> inaccessible or underground, such as conduits, cable trenches, cable troughs, duct banks, underground vaults, or direct buried in soil installations. When a power cable (greater than or equal to 480 volts) is exposed to wet, submerged, other adverse environmental conditions for which it was not designed, an aging effect of reduced insulation resistance may result, causing a decrease in the dielectric strength of the conductor insulation. This insulation degradation can be caused by wetting or submergence. This can potentially lead to failure of the cable's insulation system.

<u>Preventive Actions (XI.E3-4 and XI.E3-10)</u>
This is a condition monitoring program. However, periodic actions are taken to prevent inaccessible cables from being exposed to significant moisture, such as identifying and inspecting in-scope accessible cable conduit ends ~~and~~ <u>in</u> cable manholes for water collection, and draining the water, as needed.

The inspection frequency for water collection is established and performed based on plant-specific operating experience with cable wetting or submergence in manholes. Frequency may be adjusted based on operating experience with implemented plant design or routine actions to keep in scope cables infrequently submerged, i.e., periodic manual pump out, versus functioning drains, versus permanent automatic pumping equipment. The inspection should occur at a frequency not to exceed every two years. The inspection should include direct observation, monitoring, or indication that cables are not wetted or submerged, that cables splices are intact, and that dewatering or drainage systems (i.e., sump pumps) and associated alarms operate properly. If water is found during inspection (i.e., cable exposed to significant moisture), corrective actions are taken to keep the cable dry and ~~tests performed to~~ assess cable degradation. The first inspection for license renewal is completed prior to the period of extended operation.

Parameters Monitored/Inspected (XI.E3-5)

~~Inspection for water collection is performed based on plant-specific operating experience with water accumulation in the manhole.~~ Inaccessible or underground power (greater than or equal to 480 volts) cables within the scope of license renewal exposed to significant moisture are tested to provide an indication of the condition of the conductor insulation. The specific type of test to be used should be capable of detecting deterioration, such as reduced insulation resistance of the cable's insulation system due to wetting or submergence.

Detection of Aging Effects (XI.E3-6, 7, 8, 9, 10)

~~The inspection frequency for water collection is established and performed based on plant-specific operating experience with cable wetting or submergence in manholes (i.e., operation of dewatering devices should be inspected and operation verified prior to any known or predicted flooding events). The inspection should occur at least annually. The inspection should include direct observation that cables are not wetted or submerged, that cables/splices and cable support structures are intact, and dewatering/drainage systems (i.e., sump pumps) and associated alarms operate properly. If water is found during inspection (i.e., cable exposed to significant moisture), corrective actions are taken to keep the cable dry and tests performed to assess cable degradation. The first inspection for license renewal is completed prior to the period of extended operation.~~

Power cables exposed to significant moisture are tested at least once every ~~3 refueling cycles~~ six (6) years. This is an adequate period to monitor performance of the cable and take appropriate corrective actions since experience has shown that aging degradation is a slow process. A ~~3-refueling~~ six (6) year interval provides three to four 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 prior to the period of extended operation.

Operating Experience (XI.E3-13, 14)

Operating experience has shown that ethylene-propylene rubber (EPR) and cross-linked polyethylene (XPLE) or high molecular weight polyethylene (HMWPE) insulation materials are ~~most~~ susceptible to water tree formation. The formation and growth of water trees varies directly with operating voltage. Aging effects of reduced insulation resistance due to other mechanisms may also result in a decrease in the dielectric strength of the conductor insulation. Minimizing exposure to moisture mitigates the potential for the development of reduced insulation resistance.

~~The NRC inspectors also have continued to identify safety related cables which are submerged. The staff noted that licensees had not demonstrated that the subject safety related cables were designed for wetted or submerged service for the current license period.~~

~~Therefore, based on the operating experience, the staff revised the scope of GALL AMP XI.E3 to include all in-scope inaccessible or underground power cables at voltage levels greater than or equal to 480 volts. This ensures that all power cable conditions are monitored by inspection and testing to demonstrate that the effects of aging will be adequately managed so that the intended function(s) will be maintained consistent with the current licensing basis for the period of extended operation.~~

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## Section XI.E4, "METAL ENCLOSED BUS"

XI.E4-1 Page XI E4-1, Program Description, Paragraph 2: should clarify the details for the MEB types. Some of this information is specific to a manufacturer, and does not apply to all. The bus being insulated is not applicable to all manufacturers and applications, because this is just one way the BIL rating of the bus can be achieved during the design of the bus. For example, corona is not applicable to 480 VAC MEB applications.

XI.E4-2 Page XI E4-1, Program Description, Paragraph 3: Delete second sentence. First two sentences are redundant

XI.E4-3 Page XI E4-1, Program Description, Paragraph 5: Second sentence change "is..." to "will be...". It is presumed part of program implementation as opposed to part of the LRA Appendices A and/or B.

XI.E4-4 Page XI E4-2, Detection of Aging Effects (Element 4), Paragraph 2: third sentence, change "part of AMP's documentation" to "part of the AMP's site documentation"

XI.E4-5 Page XI E4-2, Detection of Aging Effects (Element 4), Paragraph 1: Last sentence: revise to read "Accessible elastomers (e.g., gaskets, boots, and sealants) are inspected for degradation including cracking, crazing, shrinkage, discoloration, weathering, hardening and loss of strength." Note: Corresponding line item in Table VI.A also warrants change. Consistent with elastomer degradation criteria shown in Table IX.F and the term "Hardening and loss of strength" shown in Table IX.E. Proposed wording is also more appropriate for this electrical application.

XI.E4-6 Page XI E4-3, Acceptance Criteria (Element 6), Paragraph 1: Acceptance criteria for gaskets is inconsistent with that shown in Element 4. Revise Element 6 consistent with proposal to Element 4. Inconsistency between elements. Or, simply make Element 4 read identical to Element 6. Note: Corresponding line item in Table VI.A may also warrant change.

XI.E4-7 Page XI E4-3, Corrective Actions (Element 7), Paragraph 1: 1st sentence: revise the word "required" to read "taken". Typo-consistent with Element 7 in XI.E3.

Program Description (XI.E4-1, 2, 3)

MEBs are electrical buses installed on electrically insulated supports that are constructed with each phase conductor enclosed in a separate metal enclosure (isolated phase bus), ~~or~~ all conductors enclosed in a common metal enclosure (non-segregated bus), or all phase conductors are in a common metal enclosure, but are segregated by metal barriers between phases (segregated bus). The conductors are adequately separated and insulated from ground by insulating supports or bus insulation. ~~Also, the conductors in the non-segregated bus are insulated throughout the conductor length to reduce corona and electrical tracking.~~ The MEBs are used in power systems to connect various elements in electric power circuits, such as switchgear, transformers, main generators, and diesel generators.

Industry operating experience indicates that failures of MEBs have been caused by cracked insulation and moisture or debris buildup internal to the bus duct housing. ~~Failures of MEBs have also been attributed to the cracking of bus bar insulation (bus sleeving) combined with the accumulation of moisture or debris in the bus bar enclosure.~~ Cracked insulation has resulted from high ambient temperature and contamination from bus bar joint compounds. Cracked insulation in the presence of moisture or debris has provided phase-to-phase or phase-to-ground electrical tracking paths, which has resulted in catastrophic failure of the buses. Bus failure has led to loss of power to electrical loads connected to the buses, causing subsequent reactor trips and initiating unnecessary challenges to plant systems and operators.

This AMP includes the inspection of all bus ducts within the scope of license renewal and a sample of accessible MEB bolted connections for increased resistance of connection. The technical basis for the sample selections ~~is~~ will be documented. If an unacceptable condition or situation is identified in the selected sample, a determination is made as to whether the same condition or situation is applicable to other connections not tested.

Detection of Aging Effects (XI.E4-4, 5)

MEB internal surfaces are visually inspected for aging degradation including cracks, corrosion, foreign debris, excessive dust buildup, and evidence of moisture intrusion. MEB insulating material is visually inspected for signs of embrittlement, cracking, chipping, melting, discoloration, swelling, or surface contamination. Internal bus insulating supports are visually inspected for structural integrity and signs of cracks. MEB external surfaces are visually inspected for loss of material due to general, pitting, and crevice corrosion. Accessible elastomers (e.g., gaskets, boots, and sealants) are inspected for degradation including ~~surface~~ cracking, crazing, ~~scuffing, dimensional change (e.g. "ballooning" and "necking"),~~ shrinkage, discoloration, weathering, hardening and loss of strength.

A sample of accessible bolted connections is inspected for increased resistance of connection by using thermography or by measuring connection resistance using a micro-ohmmeter. Twenty percent of the population with a maximum sample of 25 constitutes a representative sample size. Otherwise a technical justification of the methodology and sample size used for selecting components should be included as part of the AMP's site documentation. If an unacceptable condition or situation is identified in the selected sample, a determination is made as to whether the same condition or situation is applicable to other connections not tested.

Acceptance Criteria (XI.E4-6)

MEB internal surfaces show no indications of corrosion, cracks, foreign debris, excessive dust buildup, or evidence of moisture intrusion. MEB insulation materials are free from ~~regional~~ indications of surface anomalies such as embrittlement, cracking, chipping, melting, discoloration, ~~and~~ swelling, or surface contamination. ~~MEB internal surfaces show no indications of corrosion, cracks, foreign debris, excessive dust buildup, or evidence of moisture intrusion.~~

Internal bus insulating supports show no indication of structural degradation or signs of cracks. MEB external surfaces are free from loss of material due to general, pitting, and crevice corrosion. Accessible elastomers ~~gaskets, boots, and sealants~~ show no indications of cracking, crazing, discoloration, shrinkage, hardening, and loss of strength. ~~MEB external surfaces are free from loss of material due to general, pitting, and crevice corrosion.~~

Corrective Actions (XI.E4-7)

Corrective actions are ~~required~~ taken and an engineering evaluation is performed when the acceptance criteria are not met. Corrective actions may include, but are not limited, to cleaning, drying, increased inspection frequency, replacement, or repair of the affected MEB components. If an unacceptable condition or situation is identified, a determination is made as to whether the same condition or situation is applicable to other accessible or inaccessible MEBs. 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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## Section XI.E5, "FUSE HOLDERS"

XI.E5-1 Page XI E5-1, Program Description, Paragraph 1: Consider adding new pupose statement similar to the following as the 1st paragraph, "The purpose of the aging management program (AMP) described herein is to provide reasonable assurance that the intended functions of the metallic clamps of fuse holders located outside of active devices and susceptible to aging effects are maintained consistent with the current licensing basis through the period of extended operation." Or, use a reasonable facsimile. Consistency with the format of XI.E1 -XI.E4.

XI.E5-2 Page XI E5-1, Scope of Program (Element 1), Paragraph 1: 1st sentence: add "electrical transients" between thermal cycling and frequent manipulation. Consistent with corresponding line item in Table VI.A and Element 3.

XI.E5-3 Page XI E5-2, Parameters Monitored/Inspected (Element 3), Paragraph 1: Paragraph should begin "The metallic..."

XI.E5-4 Page XI E5-2, Corrective Actions (Element 7), Paragraph 1: 1st sentence: revise the word "required" to read "taken". Typo-consistent with Element 7 in XI.E3.

Program Description (XI.E5-1)

The purpose of the aging management program (AMP) described herein is to provide reasonable assurance that the intended functions of the metallic clamps of fuse holders located outside of active devices and susceptible to aging effects are maintained consistent with the current licensing basis through the period of extended operation.

Fuse holders (fuse blocks) are classified as a specialized type of terminal block because of the similarity in fuse holder design and construction to that of a terminal block. Fuse holders are typically constructed of blocks of rigid insulating material, such as phenolic resins. Metallic clamps (clips) are attached to the blocks to hold each end of the fuse. The clamps, which are typically made of copper, can be spring-loaded clips that allow the fuse ferrules or blades to slip in, or they can be bolt lugs, to which the fuse ends are bolted.

Scope of Program (XI.E5-2)
This AMP manages fuse holders (metallic clamps) located outside of active devices that are considered susceptible to the following aging effects: increased resistance of connection due to chemical contamination, corrosion, and oxidation or fatigue caused by ohmic heating, thermal cycling, electrical transients, frequent manipulation, or vibration. Fuse holders inside an active device (e.g., switchgear, power supplies, power inverters, battery chargers, and circuit boards) are not within the scope of this AMP.

Parameters Monitored/Inspected (XI.E5-3)
The metallic clamp portion of the fuse holder is tested to provide an indication of increased resistance of the connection due to chemical contamination, corrosion, and oxidation or fatigue caused by ohmic heating, thermal cycling, electrical transients, frequent manipulation or vibration.

Corrective Actions (XI.E5-4)
Corrective actions are ~~required~~ taken and an engineering evaluation is performed when the test acceptance criteria are not met in order to ensure that the intended functions of the fuse holders 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. 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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**Section XI.E6, "ELECTRICAL CABLE CONNECTIONS NOT SUBJECT TO 10 CFR 50.49 ENVIRONMENTAL QUALIFICATION REQUIREMENTS"**

XI.E6-1 Page XI E6-1, Program Description, Paragraph 1: Consider adding new purpose statement similar to the following as the 1st paragraph "The purpose of the aging management program (AMP) described herein is to provide reasonable assurance that the intended functions of the metallic parts of cable connections located outside of active devices and susceptible to age-related degradation resulting in increased resistance of connection due to thermal cycling, ohmic heating, electrical transients, vibration, chemical contamination, corrosion, or oxidation are maintained consistent with the current licensing basis through the period of extended operation." Or, use a reasonable facsimile. Consistency with the format of XI.E1 -XI.E4.

XI.E6-2 Page XI E6-2, Parameters Monitored/Inspected (Element 3), Paragraph 1: 2nd to last sentence: delete "connection type" from parameters monitored. Inconsistent with Element 3 of Final LR-ISG-2007-02 dated Dec 15, 2009. Also, the circuit application dictates the type of connection installed. Implementation of this AMP should be dictated by environmental stressors (which is the basis for all electrical AMPs) not the type of connection. If this is intended to indicate bolted connections, the addition in the first sentence is recommended.

XI.E6-3 Page XI E6-2, Parameters Monitored/Inspected (Element 3), Paragraph 1: Connection type should be deleted. This is not a contributor to stressors

XI.E6-4 Page XI E6-3, Detection of Aging Effects (Element 4), Paragraph 1: last line, change "part of AMP's documentation" to "part of the AMP's site documentation"

Program Description (XI.E6-1)

The purpose of the aging management program (AMP) described herein is to provide reasonable assurance that the intended functions of the metallic parts of bolted cable connections located outside of active devices and susceptible to age-related degradation resulting in increased resistance of connection due to thermal cycling, ohmic heating, electrical transients, vibration, chemical contamination, corrosion, or oxidation are maintained consistent with the current licensing basis through the period of extended operation.

Cable connections are used to connect cable conductors to other cable conductors or electrical devices. Connections associated with cables within the scope of license renewal are part of this aging management program (AMP). 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 AMP focuses on the metallic parts of the electrical cable connections. This AMP provides a one-time test, on a sampling basis, to ensure that either aging of metallic cable connections is not occurring and/or that the existing preventive maintenance program is effective such that a periodic inspection program is not required. The one-time test confirms the absence of age-related degradation of cable connections resulting in increased resistance of connection due to thermal cycling, ohmic heating, electrical transients, vibration, chemical contamination, corrosion, or oxidation.

Parameters Monitored/Inspected (XI.E6-2, 3)
This AMP focuses on the metallic parts of ~~the~~ bolted cable connections. The one-time testing verifies that increased resistance of connection due to thermal cycling, ohmic heating, electrical transients, vibration, chemical contamination, corrosion, or oxidation is not an aging effect that requires periodic testing. A representative sample of electrical cable connections is tested. The following factors are considered for sampling: voltage level (medium and low voltage), circuit loading (high load), ~~connection type,~~ and location (high temperature, high humidity, vibration, etc.). The technical basis for the sample selection is documented.

Detection of Aging Effects (XI.E6-4, 5)
A representative sample of electrical connections within the scope of license renewal is tested 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. Testing may include thermography, contact resistance testing, or other appropriate testing methods without removing the connection insulation, such as heat shrink tape, sleeving, insulating boots, etc. The one-time test 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. Twenty percent of the population with a maximum sample of 25 constitutes a representative sample size. Otherwise a technical

justification of the methodology and sample size used for selecting components for one-time test should be included as part of the AMP's site documentation.

**Section 2.1.3.1.3, "Regulated Events"**

2.1.3-1 Page 2.1-9, Paragraph 5, grammatical: Remove proposed change in last sentence, and change existing word "included" to "considered"

2.1.3-2 Page 2.1-9, Paragraph 5: Add new final sentence "However, the staff's review is based on the plant-specific current licensing basis, regulatory requirements, and offsite power design configurations." This is similar to Section 2.5.2.1.1.

**MARK-UP: Section 2.1.3.1.3, "Regulated Events"**

For SBO, the reviewer verifies that the applicant's methodology would include those SSCs relied upon during the "coping duration" and "recovery" phase of an SBO event. In addition, because 10 CFR 50.63(c)(1)(ii) and its associated guidance in Regulatory Guide 1.155 include procedures to recover from an SBO that include offsite and onsite power, the offsite power system that is used to connect the plant to the offsite power source should also be ~~included~~ considered within the scope of the rule. However, the staff's review is based on the plant-specific current licensing basis, regulatory requirements, and offsite power design configurations.

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**Section 2.5, " SCOPING AND SCREENING RESULTS: ELECTRICAL AND INSTRUMENTATION AND CONTROLS SYSTEMS "**

2.5-1 Page 2.5-3, Section 2.5.2.1.1: third bullet; Components Within the Scope of SBO (10 CFR 50.63) Add new final sentence "However, the staff's review is based on the plant-specific current licensing basis, regulatory requirements, and offsite power design configurations."

2.5-2 Page 2.5-3, Section 2.5.2.1.1, "Components Within the Scope of SBO (10 CFR 50.63), third bullet, third sentence, change wording from "circuit breakers" to "inter-tie devics" to keep the same terminology as the previous added sentence.

2.5-3 Page 2.5-3 and 2.5-4, Section 2.5.3, Review Procedures, 3rd paragraph, The last paragraph on page 2.5-3 that is cont. to the top of page 2.5-4 needs to be revised to clarify TLAAs associated with EQ qualificatoins. Roger provide proposed markup

2.5-4 Page 2.5-4, Section 2.5.3.1, "Components within the Scope of License Renewal," paragraph 1; First and Second lines use "component types/commodity groups" instead of "components"

## MARK-UP: Section 2.5, " SCOPING AND SCREENING RESULTS: ELECTRICAL AND INSTRUMENTATION AND CONTROLS SYSTEMS "

Page 2.5-3, Section 2.5.2.1.1

- The plant system portion of the offsite power system that is used to connect the plant to the offsite power source meeting the requirements under 10 CFR 54.4(a)(3). The electrical distribution equipment out to the first inter-tie with the offsite distribution system (i.e., equipment in the switchyard) should be included within the SBO restoration equipment scope. This path typically includes the switchyard ~~circuit breakers~~ inter-tie devices that connect to the offsite system power transformers (startup transformers), the transformers themselves, the intervening overhead or underground circuits between circuit breaker and transformer and transformer and onsite electrical distribution system, and the associated control circuits and structures. However, the staff's review is based on the plant-specific current licensing basis, regulatory requirements, and offsite power design configurations.

Page 2.5-3 and 2.5-4, Section 2.5.3

The scope of 10 CFR 50.49 electric equipment to be included within 10 CFR 54.4(a)(3) is that "long-lived" (qualified life of 40 years or greater) equipment already identified by licensees under 10 CFR 50.49(b), which specifies certain electric equipment important to safety. Licensees may rely upon their listing of environmental qualification equipment, as required by 10 CFR 50.49(d), for purposes of satisfying 10 CFR 54.4(a)(3) with respect to equipment within the scope of 10 CFR 50.49 (60 FR 22466). However, the License Renewal Rule has a requirement (10 CFR 54.21(c)) on the evaluation of TLAAs, including environmental qualification (10 CFR 50.49) analyses or calculations. Environmental qualification ~~equipment is~~ analyses are not limited to analyses for "passive" equipment. The applicant may identify environmental qualification ~~equipment~~ analyses separately for TLAA evaluation and not include ~~such~~ the equipment covered by such analyses as subject to an AMR under 10 CFR 54.21(a)(1). The environmental qualification ~~equipment~~ analyses identified for TLAA evaluation would include the "passive" environmental qualification equipment that is not subject to an AMR because it is subject to replacement based on a qualified life. The TLAA evaluation would ensure that the environmental qualification ~~equipment~~ analyses would ~~be~~ show the qualified equipment to be functional for the period of extended operation. The staff reviews the applicant's environmental qualification TLAA evaluation separately following the guidance in Section 4.4.

Page 2.5-4, Section 2.5.3.1

In this step, the staff determines whether the applicant has properly identified the ~~components~~ component types/commodity groups that are WSLR. The Rule requires that the LRA identify and list ~~components~~ component types/commodity groups that are WSLR and are subject to an AMR. Whereas, in the past, LRAs have included a table of components that are WSLR, generally that information need not be submitted with future LRAs. Although that information will be available at plant sites for inspection, the reviewer must determine through sampling of one line diagrams, and review of UFSAR and other plant documents, what portion of the components are WSLR. The reviewer must check to see if any components exist that the staff believes are within the scope but are not identified by the applicant as being subject to AMR (any request that the applicant provide justification for omitting those components that are "passive" and "long lived").

## Section 3.0, "INTRODUCTION TO STAFF REVIEW OF AGING MANAGEMENT"

3.0-1 Page 3.0-1, Section 3.0.1, "Background on the Types of Reviews", Paragraph 1, second line "with" should be replaced by "within"

3.0-2 Page 3.0-11, Table 3.0-1, "FSAR Supplement for Aging Management of Applicable Systems", Paragraph 3 (XI.E6), fifth line: add a space between "that" and "period"

3.0-3 Page 3.0-14, Table 3.0-1, Paragraph 4 (XI.E3), first line: "call" should be "calls"

3.0-4 Page 3.0-14, Table 3.0-1, Paragraph 4 (XI.E3): The frequency for this program is at least once every 5 years. The value used in GALL is 3R instead of 5 years. Neither is acceptable, but they should be consistent. The frequency of at least once every 10 years should be used with the statement that the licensee must justify this frequency. As a minimum the value of 6 years should be used. The frequency of the manhole inspections should be changed to every 2 years or reflect that annually this is a sample similar to the regional inspections for Part 50.

3.0-5 Page 3.0-16, Table 3.0-1, Paragraph 1 (XI.E2): Revise the 1st paragraph to read: "The program calls for the review of calibration results or findings of surveillance tests on electrical cables and connections used in circuits with sensitive, high-voltage, low-level current signals, such as radiation monitoring and nuclear instrumentation, to provide an indication of the existence of aging effects based on acceptance criteria related to instrumentation circuit performance. By reviewing the results obtained during normal calibration or surveillance, severe aging degradation may be detected prior to the loss of the cable and connection intended function. The review of calibration results or findings of surveillance tests is performed once every 10 years." This paragraph is confusing and technically incorrect. Normal instrument loop calibration results or surveillance tests do not provide sufficient indication to determine severe aging degradation on electrical cables and connections used in circuits with sensitive, high-voltage, low-level current signals, such as radiation monitoring and nuclear instrumentation prior to a loss of function. New paragraph as written is consistent with Element 4 of NUREG-1801, XI.E2.

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| Page 3.0-14, Table 3.0-1<br>Inaccessible Power Cables Not Subject to 10 CFR 50.49 Environmental Qualification Requirements (Chapter XI.E3) | The program call for inaccessible or underground power (greater than or equal to 480 volts) cables exposed to significant moisture to be tested at least once every ~~5~~ six (6) years to provide an indication of the condition of the conductor insulation. The specific type of test performed is determined prior to the initial test, and is to be a proven test for detecting deterioration of the insulation system due to wetting. The applicant can assess the condition of the cable insulation with reasonable confidence using one or more of the following techniques: Dielectric Loss (Dissipation Factor/Power Factor), AC Voltage Withstand, Partial Discharge, Step Voltage, Time Domain Reflectometry, Insulation Resistance and Polarization Index, Line Resonance Analysis or other testing that is state-of-the-art at the time the tests are performed. Periodic exposure to moisture for more than a few days at a time is not significant for power cables that are designed for these conditions (e.g., continuous wetting or submergence are not significant for submarine cables). In addition, inspection for water collection is established and performed based on plant-specific operating experience with water accumulation in the manholes (i.e., operation of dewatering devices should be inspected and operation verified prior to any known or predicted flooding events). However, the inspection frequency is at least ~~annual~~ once every two (2) years. | First tests or first inspections for license renewal completed prior to the period of extended operation | Electrical and Instrumentation and Control System |

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| <u>Page 3.0-16, Table 3.0-1</u><br>Insulation Material for Electrical Cables and Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements Used in Instrumentation Circuits (Chapter XI.E2) | The program calls for <u>the review of calibration results or findings of surveillance tests on</u> electrical cables and connections used in circuits with *sensitive, high-voltage, low-level current* signals, such as radiation monitoring and nuclear instrumentation, ~~to be calibrated as part of the instrumentation loop calibration at the normal calibration frequency~~ <u>to provide an indication of the existence of aging effects based on acceptance criteria related to instrumentation circuit performance</u>. ~~This calibration provides sufficient indication of the need for corrective actions based on acceptance criteria related to instrumentation loop performance.~~ <u>By reviewing the results obtained during normal calibration or surveillance, severe aging degradation may be detected prior to the loss of the cable and connection intended function</u>. The review of calibration results <u>or findings of surveillance tests</u> is performed once every 10 years.<br><br>In cases where cables are not part of calibration or surveillance program, a proven cable test (such as insulation resistance tests, time domain reflectometry tests, or other tests judged to be effective) for detecting deterioration of the insulation system are performed. The test frequency is based on engineering evaluation and is at least once every 10 years. | First review of calibration results or cable tests for license renewal completed prior to the period of *extended operation* | Electrical and Instrumentation and Control System (Section 3.6) |

## Section 3.6, " AGING MANAGEMENT OF ELECTRICAL AND INSTRUMENTATION AND CONTROLS"

3.6-1 Page 3.6-4, Section 3.6.3.2 (All): All of the subsections of 3.6.3.2 seem to be a repeat of Section 3.6.2.2. There are slight differences, but this is unnecessary repetition of information.

3.6-2 Page 3.6-8, Table 3.6-1, "Summary of Aging Management Programs for the Electrical Components Evaluated in Chapter VI of the GALL Report" (All): Order of table entries appears random. Suggest alphabetical by Component but rather the intent is to keep commodity groups together.