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XI.M9 BWR VESSEL INTERNALS

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 include information on component description and function; evaluate susceptible locations and safety consequences of failure; provide recommendations for methods, extent, and frequency of inspection; discuss acceptable methods for evaluating the structural integrity significance of flaws detected during these examinations; and recommend repair and replacement procedures.

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 (0.5 wt.% max.) steels, only static-cast steels with >20% ferrite are potentially susceptible to thermal embrittlement. Static-cast low-molybdenum steels with ≤20% ferrite and all centrifugal-cast low-molybdenum steels are not susceptible. For high-molybdenum content (2.0 to 3.0 wt.%) steels, static-cast steels with >14% ferrite and centrifugal-cast steels with >20% ferrite are potentially susceptible to thermal embrittlement. Static-cast high-molybdenum steels with ≤14% ferrite and centrifugal-cast high-molybdenum steels with ≤20% ferrite are not susceptible. In the susceptibility screening method, ferrite content is calculated by using the Hull's equivalent factors (described in NUREG/CR-4513, Rev. 1) or a method producing an equivalent level of accuracy (±6% deviation between measured and calculated values).

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

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 550° F, these materials can experience neutron embrittlement and a decrease in fracture toughness. PH-martensitic stainless steels and martensitic stainless steels also are 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 (when approved) 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>IMeV) prior to the period of extended operation, inspect five percent (5%) of the top guide locations using enhanced visual inspection technique, EVT-1 within 6 years after entering the period of extended operation. An additional 5% of the top guide locations will be inspected within 12 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 6 years after the date projected for exceeding the threshold. An additional 5% of the top guide locations are inspected within 12 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 are based on a 10% 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 TLAA analysis 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; however, maintaining high water purity reduces susceptibility to cracking due to SCC or IGSCC. Reactor coolant water chemistry is monitored and maintained in accordance with the guidelines in BWRVIP-190 (EPRI 1016579) or later revisions. 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 cracking as it affects the intended function of the component by detection and sizing of cracks through inspections 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[1]).

   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 MRP-227 guidance 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.

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

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

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 Code 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, 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-Code internals cracking is detected by using EVT-1, VT-1 or VT-3 consistent with the approved BWRVIP reports.

The applicable and approved BWRVIP guidelines recommend more stringent inspections for some internals, such as enhanced visual VT-1 (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.

Thermal and/or neutron embrittlement in susceptible CASS, PH-martensitic steels, martensitic stainless steels, and X-750 components can be 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 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 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/m^2 (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/m^2 (NUREG/CR-4513, Rev. 1).

Inspections scheduled in accordance with 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.

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

   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.

7. ***Corrective Actions:*** Repair and replacement procedures are equivalent to those requirements in ASME 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 are 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 in accordance with 10 CFR Part 50, Appendix B.

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

9. ***Administrative Controls:*** 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, 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).

Tie rod coupling has experienced IGSCC in X-750 materials 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.

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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." Adequate aging management activities for these components also includes the monitoring and control of reactor coolant water chemistry in accordance with the guidelines in EPRI 1014986, Revision 6 or later revisions to ensure long-term integrity and safe operation.

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, 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 or later revisions. 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 and loss of material by boric acid corrosion for each component. 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 invoked by the requirements of 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 program 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

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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.

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

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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.

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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.

Bulletin 2001-01, Reactor Pressure Vessel Head Degradation and Reactor Coolant Pressure Boundary Integrity, U.S. Nuclear Regulatory Commission, August 3, 2001.

Bulletin 2002-01, Reactor Pressure Vessel Head Degradation and Reactor Coolant Pressure Boundary Integrity, U.S. Nuclear Regulatory Commission, March 18, 2002.

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

NUREG-1823, *U.S. Plant Experience with Alloy 600 Cracking and Boric Acid Corrosion of Light-Water Reactor Pressure Vessel Materials*, U.S. Nuclear Regulatory Commission, April 2005.

XI.M16 PWR VESSEL INTERNALS

Program Description

This program relies on implementation of the Electric Power Research Institute (EPRI) Report No. 1016596 (MRP-227 [Ref. 1]) and EPRI Report No. 1016609 (MRP-228 [Ref. 2]) to manage the aging effects on the reactor vessel internal (RVI) components.

This program includes:

- 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 or later revisions;

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

- Monitoring and control of reactor primary coolant water chemistry in accordance with the EPRI PWR Primary Water Chemistry guidelines (EPRI Proprietary Report No. 1014986 [Ref. 3], or later revisions).

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 (1) 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; (2) loss of material induced by wear; (3) loss of fracture toughness due to either thermal aging or neutron irradiation embrittlement; (4) dimensional changes and potential loss of fracture toughness due to void swelling and irradiation growth; and (5) loss of preload due to thermal and irradiation-enhanced stress relaxation or creep. The guidance also depends 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 (Ref. 4), 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 (B&W, 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 and economically.

The result of this four-step sample selection process is a set of Primary internals 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 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 (Ref. 5, which is endorsed by reference in 10 CFR 50.55a [Ref. 6]), Examination Category B-N-3 examinations of core support structures, while a fourth set of internals locations are deemed to require No Additional Measures. As a result, the program typically inspects 5% to 15% of the RVI locations Primary Component locations for inspections, with another 7% to 10% of the RVI locations to be inspected as Expansion Components, as warranted by the evaluation of the inspection results. Another 5% to 15% 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 set 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 has incorporated into MRP-227 that clarifies how the particular visual examination methods will be used to detect relevant conditions and describes in more detail how the visual techniques related to the specific RVI components and the detection of their applicable age-related degradation effects.

The technical basis for detecting relevant conditions using volumetric ultrasonic inspection techniques can be found in MRP-228, where the review of existing bolting ultrasonic 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 ultrasonic test (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 invokes the UT criteria in MRP-228, which calls for Technical Justifications that are needed for volumetric examination method demonstrations, based on the requirements of the ASME Code, Section V (Ref. 7).

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 to be 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. pressurized water reactor (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 Babcock & Wilcox (B&W), Combustion Engineering (CE), and Westinghouse. The scope of components considered for guidance includes core support structures (typically denoted as 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. 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 also 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 (Ref. 8) 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.

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, or 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 (Ref. 9).

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: (1) cracking induced by stress corrosion cracking (SCC), primary water stress corrosion cracking (PWSCC), irradiation assisted stress corrosion cracking (IASCC), or fatigue/cyclical loading; (2) loss of material induced by wear; (3) loss of fracture toughness induced by either thermal aging, neutron irradiation embrittlement, and/or void swelling; (4) dimensional changes induced by void swelling and irradiation growth, distortion or deflection; and (5) 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 surfaces 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 boilerplates 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 boilerplates 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" (Ref. 10). No inspections, except for those specified in ASME Code Section XI, are required for components that are identified as "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: (1) 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 (2) 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 visual (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 boilerplates 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 boilerplates 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 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 locations, with the potential for inclusion of Expansion 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 Existing Programs components referenced to ASME Section XI, the IWB-3500 acceptance criteria apply. For other 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 the visual (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 boilerplate 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 guidance in MRP-227, plus the implementation of any ASME Code requirements, provides an acceptable level of aging management of safety-related components addressed in accordance with the corrective actions of 10 CFR Part 50, Appendix B (Ref. 11) or its equivalent, as applicable.

   Other alternative corrective action bases may be used to disposition relevant conditions if they have been previously approved or endorsed by the NRC. Examples of previously NRC-endorsed alternative corrective actions bases include those corrective actions bases for Westinghouse-design RVI components that are defined in Tables 4-1, 4-2, 4-3, 4-4, 4-5, 4-6, 4-7 and 4-8 of Westinghouse Report No. WCAP-14577-Rev. 1-A, (Ref. 12) or for B&W-designed RVI components in B&W Report No. BAW-2248 (Ref. 13). Westinghouse Report No. WCAP-14577-Rev. 1-A was endorsed for use in a NRC SE to the Westinghouse Owners Group, dated February 10, 2001 (Ref. 14). B&W Report No. BAW-2248 was endorsed for use in a SE to Framatome Technologies on behalf of the B&W Owners

Group, dated December 9, 1999 (Ref. 15). Alternative corrective action bases not approved or endorsed by the NRC will be submitted for NRC approval prior to their implementation.

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 or their equivalent, as applicable. It is expected that the implementation of the guidance in MRP-227 will provide an acceptable level of quality for inspection, flaw evaluation, and other elements of aging management of the PWR internals that are addressed in accordance with the 10 CFR Part 50, Appendix B or their equivalent (as applicable), confirmation process, and administrative controls.

9. ***Administrative Controls:*** The administrative controls for such programs, including their implementing procedures and review and approval processes are under existing site 10 CFR 50 Appendix B Quality Assurance Programs or their equivalent, as applicable. Such a program is thus expected to be established with a sufficient level of documentation and administrative controls to ensure effective long term implementation.

10. ***Operating Experience:*** Relatively few incidents of PWR internals aging degradation have been reported in operating U.S. commercial PWR plants. A summary of observations to date is provided in Appendix A of MRP-227-A. The applicant is expected to review subsequent operating experience for impact on its program, or participate in industry initiatives that perform this function.

    The application of the MRP-227 guidance will establish a considerable amount of operating experience over the next few years. Section 7 of MRP-227 describes the reporting requirements for these applications, and the plan for evaluating the accumulated additional operating experience.

References

1. EPRI Proprietary Report No. 1016596, "Materials Reliability Program: Pressurized Water Reactor Internals Inspection and Evaluation Guidelines (MRP-227-Rev. 0)," Electric Power Research Institute, Palo Alto, CA: 2008. (Non-publically available ADAMS Accession Number ML090160204). The non-proprietary version of the report may accessed by members of the public at ADAMS Accession Number MLXXXXXXXXX.

2. EPRI Proprietary Report No. 1016609, "Materials Reliability Program: Inspection Standard for PWR Internals (MRP-228)," Electric Power Research Institute, Palo Alto, CA [July 2009]. (Non-publically available ADAMS Accession Number ML092120574). The non-proprietary version of the report may accessed by members of the public at ADAMS Accession Number ML092750569.

3. EPRI Proprietary Report No. 1014986, 'PWR Primary Water Chemistry Guidelines, Volume 1, Revision 6," Electric Power Research Institute, Palo Alto, CA [December 2007]. (Non-publically available ADAMS Accession Number ML081140278). The non-proprietary version of the report may accessed by members of the public at ADAMS Accession Number ML081230449

4. NUREG-1800, Revision 2, "Standard Review Plan for Review of License Renewal Applications for Nuclear Power Plants," Appendix A.1, Aging Management Review - Generic (Branch Technical Position RLSB-1), U.S. Nuclear Regulatory Commission, Washington, DC [MONTH, YEAR].

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

6. Paragraph §50.55a of Title 10, *Code of Federal Regulations* (10 CFR 50.55a), "Codes and Standards," Office of the Federal Register, National Archives and Records Administration, YEAR.

7. ASME Boiler & Pressure Vessel Code, Section V, Nondestructive Examination, American Society of Mechanical Engineers, New York, NY, 2004 Edition, July 1, 2007.

8. Placeholder for Upcoming Reference on the NRC Safety Evaluation on MRP-227

9. NUREG-1801, Revision 2, "Generic Aging Lessons Learned Report," Section XI.M2, <u>Water Chemistry</u>, U.S. Nuclear Regulatory Commission, Washington, DC [MONTH YEAR].

10. NUREG-1801, Revision 2, "Generic Aging Lessons Learned Report," Section XI.M37, <u>Flux Thimble Tube Inspection</u>, U.S. Nuclear Regulatory Commission, Washington, DC [MONTH YEAR]..

11. Appendix B to Title 10, *Code of Federal Regulations* (10 CFR Part 50, Appendix B), Quality Assurance Criteria for Nuclear Power Plants, Office of the Federal Register, National Archives and Records Administration, YEAR.

12. Westinghouse Non-Proprietary Class 3 Report No. WCAP-14577-Rev. 1-A, "License Renewal Evaluation: Aging Management for Reactor Internals," Westinghouse Electric Company,

Pittsburgh, PA [March 2001]. Report was submitted to the NRC Document Control Desk in a letter dated April 9, 2001. (ADAMS Accession Number ML011080790).

13. B&W Report No. BAW-2248, "Demonstration of the Management of Aging Effects for the Reactor Vessel Internals," Framatome Technologies (now AREVA Technologies), Lynchburg VA [July 1997]. (NRC Microfiche Accession Number A0076, Microfiche Pages 001 - 108).

14. NRC Safety Evaluation from C. I. Grimes [NRC] to R. A, Newton [Chairman, Westinghouse Owners Group], "Acceptance for Referencing of Generic License Renewal Program Topical Report Entitled 'License Renewal Evaluation: Aging Management for Reactor Internals'," February 10, 2001. (ADAMS Accession Number ML010430375).

15. NRC Safety Evaluation from C. I. Grimes [NRC] to W. R. Gray [Framatome Technologies], "Acceptance for Referencing of Generic License Renewal Program Topical Report Entitled 'Demonstration of the Management of Aging Effects for the Reactor Vessel Internals,' " February 10, 2001. (ADAMS Accession Number ML993490288).

GALL Revision	WRJ Order	GallMasterID	Item	StructureAndOrComponent	Material	Environment	AgingEffect_Mechanism	AMP	FurtherEvaluation
new				PWR internals	Stainless steel; nickel alloy	Reactor coolant neutron flux	Cracking #*#due to stress corrosion cracking, irradiation-assisted stress corrosion cracking, and primary water stress corrosion cracking	Chapter XI.M2, "Water Chemistry" for PWR primary water, and #*#Chapter XI.M16A, "PWR Vessel Internals;" #*#No additional measures #*# Note: Components with no additional measures are not included in GALL tables - Components with no additional measures are defined in Section 3.3.1 of MRP-227, "Materials Reliability Program: Pressurized Water Reactor Internals Inspection and Evaluation Guidelines"	No
new				PWR internals	Stainless steel;· nickel alloy	Reactor coolant neutron flux	Loss of fracture toughness #*#due to neutron irradiation embrittlement and/or thermal embrittlement; #*#change in dimension #*#due to void swelling; #*# loss of preload #*#due to stress relaxation; #*#loss of material #*#due to wear	Chapter XI.M16A, "PWR Vessel Internals;" #*#No additional merasures #*# Note: Components with no additional measures are not included in GALL tables - Components with no additional measures are defined in Section 3.3.1 of MRP-227, "Materials Reliability Program: Pressurized Water Reactor Internals Inspection and Evaluation Guidelines"	No

new				PWR internals	Stainless steel; nickel alloy	Reactor coolant neutron flux	Cracking #*#due to stress corrosion cracking, irradiation-assisted stress corrosion cracking, and primary water stress corrosion cracking	Chapter XI.M2, "Water Chemistry" for PWR primary water, and #*#Chapter XI.M16A, "PWR Vessel Internals" #*#Inaccessible primary and expansion components	Yes, further evaluation or replacement is recommended for inaccessible primary and expansion components, if accessible primary components have defects
new				PWR internals	Stainless steel; nickel alloy	Reactor coolant neutron flux	Loss of fracture toughness #*#due to neutron irradiation embrittlement and/or thermal embrittlement; #*#change in dimension #*#due to void swelling; #*# loss of preload #*#due to stress relaxation; #*#loss of material #*#due to wear	Chapter XI.M16A, "PWR Vessel Internals" #*#Inaccessible Primary and Expansion components	Yes, further evaluation or replacement is recommended for inaccessible primary and expansion components, if accessible primary components have defects
NEW 1	133		RP-82	Core barrel assembly: #*#baffle/former assembly: #*# (a) accessible baffle-to-former bolts and screws; #*# (b) accessible locking devices (including welds) of baffle-to-former bolts and internal baffle-to-baffle bolts	Stainless steel	Reactor coolant and neutron flux	Loss of fracture toughness #*#due to neutron irradiation embrittlement; #*#change in dimension #*#due to void swelling; #*#loss of preload #*#due to stress relaxation	Chapter XI.M16A, "PWR Vessel Internals." #*#Primary components (identified in the "Structure and Components" column) #*# (for Expansion components see AMR Line Item New RP-79)	No

NEW 2	133		RP-82	Core barrel assembly: #"#baffle/former assembly; #"# (a) accessible baffle-to-former bolts and screws; #"# (b) accessible locking devices (including welds) of baffle-to-former bolts and internal baffle-to-baffle bolts	Stainless steel	Reactor coolant and neutron flux	Cracking #"#due to stress corrosion cracking, irradiation-assisted stress corrosion cracking	Chapter XI.M2, "Water Chemistry" for PWR primary water. #"#Chapter XI.M16A, "PWR Vessel Internals." #"#Primary Components (identified in the "Structure and Components" column) #"#(for Expansion components see AMR Line Item Old RP-79)	No
delete	134	1372	RP-124	Control rod guide tube assembly: #"#pipe and flange #"#spacer casting #"#rod guide tubes #"#rod guide sectors	Stainless steel	Reactor coolant and neutron flux	Cracking #"#due to stress corrosion cracking, irradiation-assisted stress corrosion cracking	Chapter XI.M2, "Water Chemistry," for PWR primary water, and Chapter XI.M16A, "PWR Vessel Internals," No Additional Measures	No
delete	135	1374	RP-126	(CRGT) assembly: #"#pipe and flange #"#spacer casting #"#spacer screws #"#flange-to-upper grid	Stainless steel	Reactor coolant and neutron flux	Changes in dimensions #"#due to void swelling	Chapter XI.M16A, "PWR Vessel Internals," No Additional Measures.	No
old	136	2285	RP-214	Control rod guide tube (CRGT) assembly: #"#accessible surfaces at four screw locations (every 90°) for CRGT spacer castings	Cast austenitic stainless steel	Reactor coolant and neutron flux	Loss of fracture toughness #"#due to thermal aging, neutron irradiation embrittlement	Chapter XI.M16A, "PWR Vessel Internals," Expansion Components (identified in the "Structure and Components" column) #"# Primary components: #"#(a) cast outlet nozzles [Oconee 3 and Davis-Besse only], 38(b) vent valve discs, and (c) Incore Monitoring Instrumentation guide tube spider castings) #"#(see AMR Line Items OLD RP-128 and NEW RP-217)	No
delete	137	1373	RP-125	(CRGT) assembly: #"#CRGT spacer screws #"#flange-to-upper grid screws #"# #"#CRGT spacer screws #"#flange-to-upper grid screws	Stainless steel	Reactor coolant and neutron flux	Cracking #"#due to stress corrosion cracking, irradiation-assisted stress corrosion cracking	Chapter XI.M2, "Water Chemistry," for PWR primary water, and Chapter XI.M16A, "PWR Vessel Internals," No Additional Measures Components.	No

delete	138	1375	RP-127	Control rod guide tube (CRGT) assembly; #*#CRGT spacer screws #*#flange-to-upper grid screws	Stainless steel	Reactor coolant and neutron flux	Loss of preload #*#due to stress relaxation	Chapter XI.M16A, "PWR Vessel Internals," No Additional Measures	No
old	139	1326	RP-79	Core barrel assembly; #*#baffle-to-former assembly; #*#(a) baffle-to-baffle bolts; (b) core barrel-to-former bolts; (c) locking devices (including welds) of external baffle-to-baffle bolts and internal baffle-to-former bolts	Stainless steel	Reactor coolant and neutron flux	Cracking #*#due to stress corrosion cracking, irradiation-assisted stress corrosion cracking	Chapter XI.M2, "Water Chemistry" for PWR primary water and #*#Chapter XI.M16A, "PWR Vessel Internals" #*#Expansion components (identified in the "Structure and Components" column) #*#(for primary components see AMR Line Item New 2 RP-82)	No
NEW	139		RP-79	Core barrel assembly; #*#baffle-to-former assembly; #*#(a) baffle-to-baffle bolts; (b) core barrel-to-former bolts; (c) locking devices (including welds) of external baffle-to-baffle bolts and internal baffle-to-former bolts	Stainless steel	Reactor coolant and neutron flux	Loss of fracture toughness #*#due to neutron irradiation embrittlement; #*#change in dimension #*#due to void swelling; #*#loss of preload #*#due to stress relaxation	Chapter XI.M16A, "PWR Vessel Internals" #*#Expansion components (identified in the "Structure and Components" column) #*#(for primary components see AMR Line Item # New 1 RP-82)	No
delete	140	1386	RP-138	#*#baffle/former assembly; #*#baffle/former bolts and screws; #*#locking devices (including	Stainless steel	Reactor coolant and neutron flux	Changes in dimensions #*#due to void swelling	Internals," #*#Expansion Components: Core barrel assembly #*# #*#Baffle/former assembly #*#Baffle/former bolts and screws #*#(for primary	No

old 1	142	1381	RP-133	Core barrel assembly: #'#accessible lower core barrel (LCB) bolts and locking devices; #'#accessible upper core barrel (UCB) bolts and locking devices	Stainless steel; nickel alloy	Reactor coolant and neutron flux	Cracking #'#due to stress corrosion cracking; irradiation-assisted stress corrosion cracking	Chapter XI.M2, "Water Chemistry," for PWR primary water, and Chapter XI.M16A, "PWR Vessel Internals" #'#Primary components (identified in the "Structure and Components" column) #'#(for Expansion components see AMR Line Items new 1 RP-133, NEW 1 RP-149, and New 3 RP-149)	No
old 2	142	1381	RP-133	Core barrel assembly: #'#accessible lower core barrel (LCB) bolts and locking devices; #'#accessible upper core barrel (UCB) bolts and locking devices	Stainless steel; nickel alloy	Reactor coolant and neutron flux	Loss of fracture toughness #'#due to neutron irradiation embrittlement	Chapter XI.M16A, "PWR Vessel Internals" #'#Primary components (identified in the "Structure and Components" column) #'#(for Expansion components see AMR Line Items New 2 RP-133, New 3 RP 141, and New 4 RP-149)	No
new 1	142	1381	RP-133	Core barrel assembly: #'#(a) upper thermal shield bolts and locking devices; (b) lower thermal shield bolts and locking divices; (c) surveillance specimen holder tube bolts and locking devices (Davis-Besse, only); (d) surveillance specimen lolder tube nuts; bolts, and locking devices (Chrystal River Unit 3, only)	Stainless steel; nickel alloy	Reactor coolant and neutron flux	Cracking #'#due to stress corrosion cracking; irradiation-assisted stress corrosion cracking	'Chapter XI.M2, "Water Chemistry," for PWR primary water, and #'#Chapter XI.M16A, "PWR Vessel Internals" #'#Expansion components (identified in the "Structure and Components" column) #'#(for Primary components see AMR Line Item old 1 RP-133)	No

new 2	142		RP-133	Core barrel assembly: #*#(a) upper thermal shield bolts and locking devices; (b) lower thermal shield bolts and locking divices; (c) surveillance specimen holder tube bolts and locking devices (Davis-Besse, only); (d) surveillance specimen lolder tube nuts; bolts, and locking devices (Chrystal River Unit 3, only)	Stainless steel; nickel alloy	Reactor coolant and neutron flux	Loss of fracture toughness #*#due to neutron irradiation embrittlement	Chapter XI.M16A, "PWR Vessel Internals" #*#Expansion components (identified in the "Structure and Components" column) #*#(for Primary components see AMR Line Item old 2 RP-133)	No
old	144	1384	RP-136	Core barrel assembly: #*#baffle plate accessible surfaces within one inch around each baffle plate flow and bolt hole	Stainless steel; nickel alloy	Reactor coolant and neutron flux	Loss of fracture toughness #*#due to neutron irradiation embrittlement	Chapter XI.M16A, "PWR Vessel Internals" Primary components (identified in the "Structure and Components" column) #*#(for expansion components see AMR Line Item new RP-136)	No
new	144	1384	RP-136	Core barrel assembly: #*#core barrel cylinder (including vertical and circumferential seam welds); #*#former plates	Stainless steel; nickel alloy	Reactor coolant and neutron flux	Loss of fracture toughness #*#due to neutron irradiation embrittlement	Chapter XI.M16A, "PWR Vessel Internals" Expansion components (identified in the "Structure and Components" column) #*#(for Primary components see AMR Line Item old RP-136)	No
delete	146	1385	RP-137	#*#core barrel cylinder (top and bottom flange); #*#lower core barrel (LCB) bolts; #*# core barrel-to- thermal shield	Stainless steel; nickel alloy	Reactor coolant and neutron flux	Loss of preload #*#due to stress relaxation	Chapter XI.M16A, "PWR Vessel Internals," No Additional Measures	No
delete	148	1379	RP-131	Core support shield (CSS) assembly: #*# #*#CSS cylinder (top flange)	Stainless steel; nickel alloy	Reactor coolant and neutron flux	Loss of fracture toughness #*#due to neutron irradiation embrittlement, void swelling	Chapter XI.M16A, "PWR Vessel Internals," No Additional Measures	No
new	148		RP-131	Core support shield (CSS) assembly: #*#CSS cylinder (top flange) #*#differential height from the top of the plenum rib pads to the reactor vessel seating surface	Stainless steel; nickel alloy	Reactor coolant and neutron flux	Loss of material #*#due to wear	Chapter XI.M16A, "PWR Vessel Internals," #*#Primary component (identified in the "Structure and Components" column) #*#No expansion components	No

old	150	1376	RP-128	Core support shield (CSS) assembly: #*#(a) CSS cast outlet nozzles (Oconee Unit 3 and Davis-Besse, only); #*#(b) CSS vent valve discs	Stainless steel; nickel alloy	Reactor coolant and neutron flux	Loss of fracture toughness #*#due to thermal aging and neutron irradiation embrittlement	Chapter XI.M16A, "PWR Vessel Internals." #*#Primary Components (identified in the "Structure and Components" column) #*#(for Expansion components see ARM Line Item old RP-214)	No
NEW			RP-128	Core support shield (CSS) assembly: #*#(a) CSS vent valve disc shafts or hinge pins; #*#(b) CSS vent valve top retaining ring: #*#(c) CSS vent valve bottom	Stainless steel; nickel alloy	Reactor coolant and neutron flux	Loss of fracture toughness #*#due to thermal aging and neutron irradiation embrittlement	Chapter XI.M16A, "PWR Vessel Internals." #*#Primary Components (identified in the "Structure and Components" column) #*# No expansion components	
delete	152	1377	RP-129	Core support shield (CSS) assembly: #*#CSS cylinder (top and bottom flange) #*#upper core barrel (UCB) bolts and their locking devices	Stainless steel; nickel alloy	Reactor coolant	Cracking #*#due to stress corrosion cracking, primary water stress corrosion cracking, irradiation-assisted stress corrosion cracking	Chapter XI.M2, "Water Chemistry," for PWR primary water, and Chapter XI.M16A, "PWR Vessel Internals," #*#Primary Components: Accessible upper core barrel bolts; accessible upper core	No
delete	154	1393	RP-145	Flow distributor assembly: #*#flow distributor head and flange; #*#incore guide support plate; #*#clamping ring;	Stainless steel; nickel alloy	Reactor coolant	Cracking #*#due to stress corrosion cracking, irradiation-assisted stress corrosion cracking	Chapter XI.M2, "Water Chemistry," for PWR primary water, and Chapter XI.M16A, "PWR Vessel Internals," expansion components. #*#Expansion Components:	No
delete	171	1396	RP-148	Flow distributor assembly: #*#flow distributor head and flange #*#shell forging-to-flow distributor bolts #*#incore guide support	Stainless steel; nickel alloy	Reactor coolant and neutron flux	Loss of fracture toughness #*#due to neutron irradiation embrittlement, void swelling	Chapter XI.M16A, "PWR Vessel Internals," No Additional Measures. #*#No Additional Measures: All of the components of the flow distributor assembly for the	No
delete	173	1397	RP-149	Flow distributor assembly #*# #*#Clamping ring #*# #*#Shell forging-to- flow distributor bolts	Stainless steel; nickel alloy	Reactor coolant and neutron flux	Loss of preload #*#due to stress relaxation	Chapter XI.M16A, "PWR Vessel Internals," No Additional Measures.	No

new 1	173		RP-149	Flow distributor assembly: #*#lower grid shock pad bolts and locking devices (TMI -1, only)	Stainless steel; nickel alloy	Reactor coolant and neutron flux	Cracking #*#due to stress corrosion cracking, irradiation-assisted stress corrosion cracking	Chapter XI.M2, "Water Chemistry," for PWR primary water, and Chapter XI.M16A, "PWR Vessel Internals," Expansion components (identified in the "Structure and Components" column) #*#(for Primary components see ARM Line Item Old 1 RP-133)	No	
new 2	173		RP-149	Flow distributor assembly: #*#lower grid shock pad bolts and locking devices (TMI -1, only)	Stainless steel; nickel alloy	Reactor coolant and neutron flux	Loss of fracture toughness #*#due to neutron irradiation embrittlement, void swelling	Chapter XI.M16A, "PWR Vessel Internals" #*#Expansion components (identified in the "Structure and Components" column) #*#(for Primary components see ARM Line Item old 2 RP-133)	No	
new 3			RP-149	Flow distributor assembly: #*#flow distributor bolts and locking devices	Stainless steel; nickel alloy	Reactor coolant and neutron flux	Cracking #*#due to stress corrosion cracking, irradiation-assisted stress corrosion cracking	Chapter XI.M2, "Water Chemistry," for PWR primary water, and Chapter XI.M16A, "PWR Vessel Internals," Expansion components (identified in the "Structure and Components" column) #*# (for Primary components see ARM Line Item Old 1 RP-133)	No	
new 4			RP-149	Flow distributor assembly: #*#flow distributor bolts and locking devices	Stainless steel; nickel alloy	Reactor coolant and neutron flux	Loss of fracture toughness #*#due to neutron irradiation embrittlement, void swelling	Chapter XI.M16A, "PWR Vessel Internals" #*#Expansion components (identified in the "Structure and Components" column) #*#(for Primary components see ARM Line Item old 2 RP-133)	No	
NEW	175	2288	RP-217	Incore Monitoring Instrumentation (IMI) guide tube assembly: #*#accessible IMI Incore guide tube spider castings	Cast austenitic stainless steel	Reactor coolant and neutron flux	Loss of fracture toughness #*#due to thermal aging, neutron irradiation embrittlement	Chapter XI.M16A, "PWR Vessel Internals" #*#Primary components (identified in the "Structure and Components" column) #*#(for Expansion components see Line Item old RP-142)	No	

Old			RP-217	'Incore Monitoring Instrumentation (IMI) guide tube assembly: #"#IMI guido tube spider-to-lower grid rib section welds	Cast austenitic stainless steel	Reactor coolant and neutron flux	Loss of fracture toughness #"#due to thermal aging, neutron irradiation embrittlement	Chapter XI.M16A, "PWR Vessel Internals" #"#Primary components (identified in the "Structure and Components" column) #"#(for expansion components see Line Item old RP-142)	No
old	177	1390	RP-142	Lower grid assembly: #"#(a) accessible pads; #"#(b) pad-to-rib section welds; #"#(c) alloy X-750 dowels, cap screws and locking devices	Stainless steel; nickel alloy	Reactor coolant and neutron flux	Loss of fracture toughness #"#due to neutron irradiation embrittlement	Chapter XI.M16A, "PWR Vessel Internals" #"#Expansion components (identified in the "Structure and Components" column) #"#(for primary componetns see AMR Line Items New and New PR-217)	No
old	179	1389	RP-141	Lower grid assembly: #"#alloy X-750 dowel-to-guide block welds	Stainless steel; nickel alloy	Reactor coolant and neutron flux	Cracking #"#due to stress corrosion cracking, primary water stress corrosion cracking, irradiation-assisted stress corrosion cracking	Chapter XI.M2, "Water Chemistry," for PWR primary water, and Chapter XI.M16A, "PWR Vessel Internals" #"#Primary components (identified in the "Structure and Components" column) #"#(for Expansion components see AMR Line Item new 1 RP-141)	No
new 1			RP-141	Lower grid assembly: #"#accessible alloy X-750 dowel locking welds to the upper and lower fuel assembly support pads	Stainless steel; nickel alloy	Reactor coolant and neutron flux	Cracking #"#due to stress corrosion cracking, primary water stress corrosion cracking, irradiation-assisted stress corrosion cracking	'Chapter XI.M2, "Water Chemistry," for PWR primary water, and Chapter XI.M16A, "PWR Vessel Internals" #"#Expansion components (identified in the "Structure and Components" column) #"#(for primary components see AMR Line Item old RP-141)	No

new 2	179	1389	RP-141	Lower grid assembly: #*#lower grid shock pad bolts and locking devices (TMI-1, only)	Stainless steel; nickel alloy	Reactor coolant and neutron flux	Cracking #*#due to stress corrosion cracking, primary water stress corrosion cracking, irradiation-assisted stress corrosion cracking	Chemistry," for PWR primary water, and Chapter XI.M16A, "PWR Vessel Internals" #*#Expansion components (identified in the "Structure and Components" column) #*#(for primary components see AMR Line Item old 1 RP-133)	No
new 3			RP-141	Lower grid assembly: #*#lower gris shock pad bolts and locking devices (TMI-1, only)	Stainless steel; nickel alloy	Reactor coolant and neutron flux	Loss of fracture toughness #*#due to neutron irradiation embrittlement	Chapter XI.M16A, "PWR Vessel Internals" #*#Expansion components (identified in the "Structure and Components" column) #*#(for Primary components see AMR Line Item old 2 RP-133)	
delete	181	1365	RP-117	Plenum cover assembly: #*#plenum cylinder; #*#reinforcing plates; #*#top flange-to-cover bolts; #*#bottom	Stainless steel	Reactor coolant	Cracking #*#due to stress corrosion cracking, irradiation-assisted stress corrosion cracking	Internals," No Additional Measures. #*#No Additional Measures: The top flange-to-cover bolts, the plenum cover assembly, the plenum cylinder,	No
delete	183	1366	RP-118	plenum cylinder #*#Plenum cover assembly: #*#plenum cylinder; #*#reinforcing	Stainless steel	Reactor coolant	Cracking #*#due to stress corrosion cracking, irradiation-assisted stress corrosion cracking	Internals," No Additional Measures. #*#No Additional Measures: The top flange-to-cover bolts, the plenum cover	No
delete	185	1006	RP-24	Reactor vessel internal components	Stainless steel; nickel alloy	Reactor coolant	Loss of material #*#due to pitting and crevice corrosion	Chapter XI.M2, "Water Chemistry" for PWR primary water	No
delete	187	1398	RP-150	Core barrel assembly #*# #*#Thermal shield cylinder	Stainless steel	Reactor coolant	corrosion cracking, irradiation-assisted stress corrosion cracking	Chemistry," for PWR primary water, and Chapter XI.M16A, "PWR Vessel Internals," No	No
Rev 02	189	2284	RP-213	Plenum cover and plenum cylinder assemblies: #*#(a) plenum rib pads (weldment rib pads) and (b) support flange #*#Differential height between top of plenum rib pads and reactor vessel seating surface, with plenum in vessel, for wear	Stainless steel	Reactor coolant and neutron flux	Loss of material and associated loss of clamping load #*#due to wear	Chapter XI.M16A, "PWR Vessel Internals." #*#Primary components (identified in the "Structure and Components" column) #*#No Expansion components	No

delete	191	1368	RP-120	Plenum cover and plenum cylinder assemblies: #*#plenum rib pads (Weldment rib pads) #*#upper grid assembly #*#rib section	Stainless steel	Reactor coolant	Cracking #*#due to stress corrosion cracking, irradiation-assisted stress corrosion cracking	Chapter XI.M2, "Water Chemistry," for PWR primary water, and Chapter XI.M16A, "PWR Vessel Internals," No Additional Measures. #*#No Additional Measures: All upper	No
delete	193	1371	RP-123	Plenum cover and plenum cylinder assemblies: #*#plenum rib pads (Weldment rib	Stainless steel	Reactor coolant and neutron flux	Loss of fracture toughness #*#due to neutron irradiation embrittlement, void swelling	Chapter XI.M16A, "PWR Vessel Internals," No Additional Measures. #*#No Additional Measures: All plenum cover,	No
delete	139	1326	RP-79	Core barrel assembly #*# #*#Baffle/former assembly #*#Baffle/former bolts and screws	Stainless steel	Reactor coolant and neutron flux	Cracking/ stress corrosion cracking, irradiation-assisted stress corrosion cracking	Internals." #*#Primary Components: Accessible baffle/former bolts and screws; Accessible baffle-to-former and internal baffle-to-baffle bolt locking devices #*#(for	No
delete	141	1387	RP-139	#*#Baffle/former assembly; #*#Baffle/former bolts and screws; #*#Locking devices (including welds) of baffle/former	Stainless steel	Reactor coolant	Loss of preload #*#due to stress relaxation	Internals." #*#Expansion Components: Core barrel assembly #*# #*#Baffle/former assembly #*#Baffle/former bolts and screws #*#(for primary components see AMR Line Item	No

RIC 2010 – License Renewal Program

Wednesday, March 10, 10:30 AM – 12:30 PM

Presenter	Mins	Topic
• Brian Holian	10	Introductions and Opening
• Jerry Dozier	10	LR Guidance Documents Update
• Andy Imboden	10	LR Environmental Guidance Documents Update
• Rich Conte	15	Inspection Insights Related to IP71003
• Ann Marie Stone	10	Inspection Insights & Assessing & Documenting Concerns
• Garry Young, NEI	15	NEI Perspectives
• Toru Osaki, JNESO	15	Degradation of Materials of Components & Structures in Early-Constructed Japanese LWRs
• Brian Holian/All	35	Question/Answer & Summary/Close Out