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# SLR-ISG-PWRVI-2020-XX

# Updated Aging Management Criteria for Reactor Vessel Internal Components for Pressurized-Water Reactors

## Draft Interim Staff Guidance

## July 2020

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# Draft Interim Staff Guidance

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**DRAFT INTERIM STAFF GUIDANCE**

# UPDATED AGING MANAGEMENT CRITERIA FOR REACTOR VESSEL INTERNAL COMPONENTS FOR PRESSURIZED-WATER REACTORS SUBSEQUENT LICENSE RENEWAL GUIDANCE

**SLR-ISG-PWRVI-2020-XX**

## PURPOSE

The U.S. Nuclear Regulatory Commission (NRC) staff is issuing this draft subsequent license renewal (SLR) interim staff guidance (ISG) to provide clarifying guidance to facilitate staff and industry understanding of the aging management of systems, structures, and components required by Title 10 of the Code of Federal Regulations (10 CFR) Part 54, "Requirements for renewal of operating licenses for nuclear power plants" (Ref. 1).

This draft SLR-ISG identifies proposed revisions to the guidance for pressurized-water reactor (PWR) vessel internal components in NUREG 2192, "Standard Review Plan for Review of Subsequent License Renewal Applications for Nuclear Power Plants" (SRP-SLR), issued July 2017 (Ref. 2), and in NUREG-2191, Volumes 1 and 2, "Generic Aging Lessons Learned for Subsequent License Renewal (GALL SLR) Report," issued July 2017 (Ref. 3).

The guidance in this SLR-ISG supersedes in total the previous guidance in License Renewal (LR)-ISG-2011-04, "Updated Aging Management Criteria for Reactor Vessel Internal Components of Pressurized Water Reactors," dated June 3, 2013 (Ref. 4), which is related to NUREG-1801, Revision 2, "Generic Aging Lessons Learned (GALL) Report," issued December 2010 (Ref. 5), and NUREG 1800, Revision 2, "Standard Review Plan for Review of License Renewal Applications for Nuclear Power Plants" (SRP-LR), issued December 2010 (Ref. 6).

## BACKGROUND

The NRC staff has reviewed three applications to extend plant operations to 80 years (i.e., for SLR) for Turkey Point Nuclear Generating Units 3 and 4 (Turkey Point); Peach Bottom Atomic Power Station, Units 2 and 3 (Peach Bottom); and Surry Power Station, Units 1 and 2 (Surry). During these reviews, both the staff and applicants have identified ways to make the preparation and review of future subsequent license renewal applications (SLRAs) more effective and efficient.

## RATIONALE

Public meetings took place on March 28, 2019; December 12, 2019; February 20, 2020; March 25, 2020; April 3, 2020; and April 7, 2020, between the staff and industry representatives to discuss staff and industry experience in the preparation and review of the initial license renewal application (LRA) for River Bend Station, Unit 1, which piloted the optimized 18-month review process for SLRAs, as well as the reviews of the first three SLRAs for Turkey Point, Peach Bottom, and Surry.

The guidance document changes issued in this SLR-ISG are based on the updated inspection and evaluation (I&E) guidelines in Electric Power Research Institute (EPRI) Materials Reliability

Program (MRP) Topical Report No. 3002017168, "Materials Reliability Program: Pressurized Water Reactor Internals Inspection and Evaluation Guidelines (MRP-227, Revision 1-A)," issued December 2019 (Ref. 7), which the NRC staff found acceptable for referencing in licensing applications in its safety evaluation dated April 25, 2019 (Ref. 8), and approved for use in the staff's letters to the EPRI MRP dated February 19, 2020 (Ref. 9), and July 7, 2020 (Ref. 10).

The NRC is issuing this draft SLR-ISG to accomplish the following five objectives:

(1) <u>GALL-SLR Report and SRP-SLR Guidance Changes</u>: Update the staff's guidance for PWR reactor vessel internal (RVI) components in the GALL-SLR Report and SRP-SLR to account for changes in I&E criteria for PWR RVI components made in MRP-227, Revision 1-A, and in other relevant industry documents (e.g., EPRI MRP expert panel reports for 80-year RVI component assessments or in relevant industry interim guidance documents or alert letters).

(2) <u>Clarification on the Use of MRP-227, Revision 1-A</u>: Clarify whether incorporation and adoption of MRP-227, Revision 1-A, may be used as the starting basis for the PWR Vessel Internals Aging Management Program (AMP) and whether reference to the criteria in MRP-227, Revision 1-A, in a PWR applicant's SLRA will need to be subject to the performance of an RVI component-specific gap analysis.

(3) <u>Reduction of Unnecessary Burden for PWR SLRAs</u>: Provide additional clarifications on PWR Vessel Internals AMP programmatic change bases that are considered to be administrative and that will no longer need to be within the scope of AMP-identified exceptions or enhancements.

(4) <u>Resolution of Applicant/Licensee Action Items (A/LAIs)</u>: Resolve whether the staff's A/LAIs in its safety evaluation for the I&E guidelines in EPRI TR No. 1022863, "Materials Reliability Program: Pressurized Water Reactor Internals Inspection and Evaluation Guidelines (MRP-227-A)," dated December 16, 2011 (Ref. 11), and A/LAI No. 1 in the staff's safety evaluation for the I&E guidelines in MRP-227, Revision 1-A, dated April 25, 2019, need to be addressed in an initial LRA or an SLRA.

(5) <u>Closure of Regulatory Information Summary (RIS) 2011-07</u>: Provide the staff's basis for closing previous guidance matters raised in RIS 2011-07, "License Renewal Submittal Information for Pressurized Water Reactor Internals Aging Management," dated July 21, 2011 (Ref. 12).

## CURRENT REGULATORY FRAMEWORK

The NRC defines and establishes the staff's rules for submitting and receiving Commission approval of LRAs or SLRAs in 10 CFR Part 54. Pursuant to the requirements specified in 10 CFR 54.21(a)(1), a license renewal applicant is required to perform an integrated plant assessment of its facility to determine those systems, structures, or components (SSCs) that are within the scope of an aging management review (AMR). In 10 CFR 54.21(a)(1), the NRC defines SSCs subject to an AMR as those SSCs that perform an intended function in accordance with the requirements defined in 10 CFR 54.4, "Scope," without moving parts or a change in configuration, and that are not subject to replacement based on a qualified life or specified time period (sometimes referred to as "passive, long-lived" components). For those SSCs that are within the scope of an AMR, 10 CFR 54.21(a)(3) requires the applicant to demonstrate that the effects of aging on the SSCs 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.

The requirements in 10 CFR 54.21(a)(1) and 10 CFR 54.21(a)(3) apply to subsequent periods of extended operation that may be proposed in an SLRA for a U.S light-water reactor facility. The PWR RVI components that are within the scope of this SLR-ISG are those that are required to be the subject of an AMR pursuant to the integrated plant assessment requirements in 10 CFR 54.21(a)(1).

The guidance in this SLR-ISG provides a process that may be used to determine whether a specified PWR RVI component will need to be managed for specified aging effects in accordance with the requirements defined in 10 CFR 54.21(a)(3).

## DISCUSSION

AMP XI.M16A, "PWR Vessel Internals," of the GALL Report, Revision 2, and the associated AMR line items in both the GALL Report, Revision 2, and SRP-LR, Revision 2, provide aging management guidance for PWR vessel internals based on the initial submitted version of MRP-227, Revision 0, dated December 2008 (Ref. 13). LR-ISG-2011-04 updated GALL Report Revision 2 AMP XI.M16A to be consistent with MRP-227-A (Ref. 14), which the NRC staff approved in a safety evaluation dated December 16, 2011 (Ref. 11). The staff also updated the AMR line items for PWR RVI components in both the GALL Report, Revision 2, and SRP-LR, Revision 2, to make them consistent with MRP-227-A.

The NRC issued the GALL-SLR Report and SRP-SLR in 2017 to address plant operation for a period up to 80 years. The AMR line items were based on those provided in LR-ISG-2011-04, as adjusted for relevant operating experience or industry recommendations that were developed after the issuance of MRP-227-A. However, these AMR line items did not represent a complete analysis for 80 years of operations.

GALL-SLR Report AMP XI.M16A and SRP-SLR Section 3.1.2.2.9 were based on MRP-227-A, which is an analysis for 60 years of plant operation. These GALL-SLR Report and SRP-SLR sections used the term "MRP-227-A (as supplemented)" to describe either the use of MRP-227-A as supplemented by a gap analysis to enhance the program for an 80-year operating period, or the use of acceptable generic guidance such as an approved revision of MRP-227 that considers an operating period of 80 years. For example, in SRP-SLR Section 3.1.2.2.9, the staff clarified that if a gap analysis is needed for the programmatic basis, the analysis should consider the extension of time-dependent cyclical loads and neutron irradiation exposures through the end of an 80-year cumulative licensing period to identify changes to inspections of PWR RVI components from those defined for the specified components in MRP-227-A. The staff also explained that an SLRA does not need to include a gap analysis of the RVI components if the AMP is based on a site-specific or staff-approved generic industry program whose evaluation of aging in the RVI components is based on an 80-year assessment.

The revisions in this SLR-ISG to the information for PWR RVI components in the GALL-SLR Report and SRP-SLR reflect the revised I&E guidelines in MRP-227, Revision 1-A. While Revision 1-A is an update of the guidance in MRP-227-A that reflects the operating experience since the issuance of MRP-227-A, Revision 1-A only assesses PWR RVI components through the end of a 60-year licensing term. Thus, even if an applicant revises its PWR vessel internals program (or analogous AMP for the RVI components) based on MRP-227, Revision 1-A, the program in the SLRA will need a gap analysis to identify enhancements to the program that are necessary to address an 80-year operating period. As described in SRP-SLR Section 3.1.2.2.9 (as updated in this SLR-ISG), the SLRA should include and discuss the gap analysis methods and results. As a result of these considerations, the staff considers that it is appropriate to issue this SLR-ISG that covers updated aging management criteria and bases for PWR RVI components.

**APPLICABILITY**

All holders of operating licenses for nuclear power reactors under 10 CFR Part 50, "Domestic licensing of production and utilization facilities" (Ref. 15), except those that have permanently ceased operations and have certified that fuel has been permanently removed from the reactor vessel.

**GUIDANCE**

The NRC provides requirements for the submission and review of applications to extend plant operations beyond the initial 40-year operating period in 10 CFR Part 54.

The GALL-SLR Report and SRP-SLR provide guidance to licensees that wish to extend their plant operating licenses from 60 years to 80 years, and to the NRC staff who will review the SLRAs.

The staff and nuclear industry have identified a number of areas for which future SLRAs and staff reviews can be completed more effectively and efficiently. A series of SLR-ISGs captures these areas, known as lessons learned.

The NRC staff considers that the information in this ISG provides an acceptable approach for managing aging in PWR vessel internal components within the scope of 10 CFR Part 54 and will improve the quality, uniformity, effectiveness, and efficiency of NRC staff reviews of future SLRAs.

**IMPLEMENTATION**

The NRC staff will use the information discussed in this draft SLR-ISG to determine whether, pursuant to 10 CFR 54.21(a)(3), an SLRA demonstrates that the effects of aging on structures and components subject to an AMR are adequately managed so their intended functions will be maintained consistent with the current licensing basis for the subsequent period of extended operation. This draft ISG contains an update in redline/strikeout of the GALL-SLR Report and SRP-SLR sections related to the aging management of pressurized-water RVIs. An applicant may reference this SLR-ISG in an SLRA to demonstrate that the AMPs at the applicant's facility correspond to those described in the GALL-SLR Report. If an applicant credits an AMP as updated by this ISG, it is incumbent on the applicant to ensure that the conditions and operating experience at the plant are bounded by the conditions and operating experience for which this draft ISG was evaluated. If these bounding conditions are not met, it is incumbent on the applicant to address any additional aging effects and augment its AMPs.

For AMPs that are based on this ISG, the NRC staff will review and verify whether the applicant's AMPs are consistent with those described in this ISG, including applicable plant conditions and operating experience.

**ACTIONS**

SLR-ISG Objectives 1 and 2—GALL-SLR Report and SRP-SLR Guidance Changes and Clarifications on the Use of MRP-227, Revision 1-A

This SLR-ISG updates the following sections or tables in the GALL-SLR Report or SRP-SLR to ensure consistency with guidance in MRP-227, Revision 1-A:

- commodity group-based AMR line items for PWR RVI components in Table 3.1-1 of the SRP-SLR
- AMR line items for these components in Table IV.B2 of the GALL-SLR Report
- AMR line items for these components in Table IV.B3 of the GALL-SLR Report
- AMR line items for these components in Table IV.B4 of the GALL-SLR Report
- generic AMR line items applying to PWR RVI components in Section IV.E and Table IV.E of the GALL-SLR Report
- AMR further evaluation acceptance criteria for PWR RVI components in SRP-SLR Section 3.1.2.2.9 and AMR further evaluation review procedures for PWR RVI components in SRP-SLR Section 3.1.3.2.9
- the program description, program elements, and program references in GALL-SLR Report AMP XI.M16A
- the final safety analysis report (FSAR) supplement example for a PWR vessel internals program specified in Table 3.0-1 of the SRP-SLR
- material definitions in GALL-SLR Report Table IX.C to add a new definition for stellite materials, which may apply to the design of specific types of PWR RVI components
- SRP-SLR Table 4.7-1 to include MRP-based fluence and cycle analyses for PWR RVI components as potential plant-specific time-limited aging analyses (TLAAs) for PWR SLRAs

The appendices included in this SLR-ISG provide the updated versions of these sections, line items, or tables.

MRP-227, Revision 1-A, is based (in part) on an assessment of Westinghouse, Combustion Engineering (CE), and Babcock & Wilcox (B&W)-designed reactor internals over a 60-year cumulative licensed service life for the reactors. Thus, PWR SLR applicants who transition their programs to use MRP-227, Revision 1-A as the starting basis for their AMPs, will need to perform and include a gap analysis for their PWR RVI components in their SLRAs.

These actions satisfy Objectives 1 and 2, as stated in the Rationale section of this SLR-ISG.

SLR-ISG Objective 3—Reduction of Unnecessary Burden for PWR SLRAs

The PWR Vessel Internals program discussed in this SLR-ISG is based on MRP-227, Revision 1-A. The programs in SLRAs may also include implementation of additional inspection guidance developed by the EPRI MRP, industry vendors, or owners' organizations (e.g., Westinghouse, CE, B&W, or the PWR Owners Group). The NRC has updated the "Scope of Program" element in GALL-SLR Report AMP XI.M16A to clarify that the scope of PWR vessel internals programs may include all industry guidelines that apply to the RVI components. The "Administrative Controls" and "Confirmation Process" elements in GALL-SLR Report AMP XI.M16A identify that the program implements these guidelines in accordance with an applicant's industry initiative processes in accordance with Nuclear Energy Institute 03-08, "Guideline for the Management of Materials Issues," Revision 3, dated February 2017 (Ref. 16). The staff acknowledges that, as the industry generates supplemental guidance, the plant procedures for these programs may not be up to date with the new methods recommended for the components. Activities to update and maintain the procedures are explicitly identified in the "Confirmatory Processes" and "Administrative Controls" elements of the AMP.

These clarifications satisfy Objective 3, as stated in the Rationale section of this SLR-ISG.

SLR-ISG Objective 4—Resolution of A/LAIs

The safety evaluation for MRP-227-A identified a number of A/LAIs to be addressed by those applicants or licensees using that topical report to satisfy the aging management requirements of 10 CFR 54.21(a)(3).

The staff's approval basis in the April 25, 2019, safety evaluation for MRP-227, Revision 1-A, was sufficient to close all A/LAIs previously issued by the staff on MRP-227-A. Therefore, responses to the A/LAIs on MRP-227-A do not need to be included in a PWR SLRA or in a PWR LRA where the PWR vessel internals program for the SLRA or LRA is based on the I&E guidelines in MRP-227, Revision 1-A.

The safety evaluation for MRP-227, Revision 1-A, did identify one A/LAI, which pertains to an applicant's basis for resolving generic operating experience with the occurrence of cracking in Westinghouse-designed baffle-former bolts or CE-designed core shroud bolts. Since A/LAI No. 1 on MRP-227, Revision 1-A, is applicable to an SLR applicant's basis for addressing relevant operating experience, it is acceptable for the applicant to address its resolution of A/LAI No. 1 as part of its bases for addressing relevant operating experience for the baffle-former bolts or core shroud bolts in the "Operating Experience" program element of the applicant's PWR Vessel Internals AMP, or in the applicant's technical basis document or procedure for the AMP. A separate SLRA section addressing the A/LAI would not be necessary. The clarifications made in this Actions section satisfy Objective 4, as referenced in the Rationale section of this SLR-ISG.

SLR-ISG Objective 5—Closure of RIS 2011-07

The staff's guidance in RIS 2011-07 addresses differences in aging management criteria for a plant's PWR RVI components based on the timing of the initial LRA submittal and the applicability and specified guidance criteria in the GALL Report version referenced in the LRA. The guidance in RIS 2011-07 no longer applies to future license renewal or SLR applicants because LRAs will be submitted in accordance with the criteria in either the GALL-SLR Report or the GALL Report, Revision 2, and SLRAs will be submitted in accordance with the GALL-SLR

Report. As such, the staff is formally closing the guidance of RIS 2011-07 in SLR-ISG-PWRVI-2020-XX.

The clarification made in this Actions section satisfies Objective 5, as referenced in the Rationale section of this SLR-ISG.

**NEWLY IDENTIFIED SYSTEMS, STRUCTURES, AND COMPONENTS UNDER 10 CFR 54.37(b)**

Any structures and components identified in this SLR-ISG as requiring aging management that were not previously identified in earlier versions of the SRP-SLR or GALL-SLR Report are considered to be newly identified structures and components under 10 CFR 54.37(b). Specifically, the staff's update of AMR items and GALL-SLR Report AMP XI.M16A in this SLR-ISG is based (in part) on the EPRI MRP's analysis of PWR RVI components in MRP-227, Revision 1-A. Any new components identified for aging management in this SLR-ISG are based on the EPRI MRP's analysis and decision to place new PWR RVI components in the "Primary," "Expansion," or "Existing Program" categories of MRP-227, Revision 1-A, in addition to those that these categories previously included in MRP-227-A.

**BACKFITTING AND ISSUE FINALITY DISCUSSION**

Discussion to be provided in the final ISG.

**CONGRESSIONAL REVIEW ACT**

Discussion to be provided in the final ISG.

**FINAL RESOLUTION**

By July 1, 2027, the staff will transition this information into NUREG-2191 (GALL-SLR Report) and NUREG-2192 (SRP-SLR). Following the transition of this guidance to NUREG-2191 and NUREG-2192, this ISG will be closed.

**APPENDICES**

A. Proposed Revisions to SRP-SLR Table 3.1-1

B.1 Proposed Revisions to GALL-SLR Report Table IV.B2, "Reactor Vessel Internals (PWR)—Westinghouse"

B.2 Proposed Revisions to GALL-SLR Report Table IV.B3, "Reactor Vessel Internals (PWR)—Combustion Engineering"

B.3 Proposed Revisions to GALL-SLR Report Table IV.B4, "Reactor Vessel Internals (PWR)—Babcock & Wilcox"

B.4 Proposed Revisions to GALL-SLR Report Table IV.E, "Common Miscellaneous Material/Environment Combinations"

C. Proposed Revisions to SRP-SLR Section 3.1.2.2.9, (AMR Further Evaluation Acceptance Criteria) and SRP-SLR Section 3.1.3.2.9 (AMR Further Evaluation Review Procedures)

D. Proposed Revisions to GALL-SLR Report AMP XI.M16A, “PWR Vessel Internals,” and Related FSAR Supplement Example in GALL-SLR Report Table XI-01

E. Proposed Revision to GALL-SLR Report Table IX.C, “Use of Terms for Materials”

F. Proposed Revisions to SRP-SLR Table 4.7-1, “Examples of Potential Plant-Specific TLAA Topics”

G. List of Abbreviations Commonly Used in SLR-ISG-PWRVI-2020-XX

**REFERENCES**


1. *U.S. Code of Federal Regulations*, “Requirements for renewal of operating licenses for nuclear power plants,” Part 54, Chapter 1, Title 10, “Energy.”

2. NUREG-2192, “Standard Review Plan for Review of Subsequent License Renewal Applications for Nuclear Power Plants,” July 2017 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML17188A158).

3. NUREG-2191, Volumes 1 and 2, “Generic Aging Lessons Learned for Subsequent License Renewal (GALL-SLR) Report,” July 2017 (ADAMS Accession Nos. ML17187A031 and ML17187A204).

4. NRC Interim Staff Guidance LR-ISG-2011-04, “Updated Aging Management Criteria for Reactor Vessel Internal Components for Pressurized Water Reactors,” June 3, 2013 (ADAMS Accession No. ML12270A436).

5. NUREG-1801, Revision 2, “Generic Aging Lessons Learned (GALL) Report,” December 2010 (ADAMS Accession No. ML103490041).

6. NUREG-1800, Revision 2, “Standard Review Plan for Review of License Renewal Applications for Nuclear Power Plants,” December 2010 (ADAMS Accession No. ML103490036).

7. EPRI Technical Report No. 3002017168, “Materials Reliability Program: Pressurized Water Reactor Internals Inspection and Evaluation Guidelines (MRP-227, Revision 1-A),” December 2019 (ADAMS Accession No. ML19339G350).

8. NRC Safety Evaluation, “Final Safety Evaluation for Electric Power Research Institute Topical Report MRP-227, Revision 1, ‘Materials Reliability Program: Pressurized Water Reactor Internals Inspection and Evaluations Guideline,’” April 25, 2019 (ADAMS Accession No. ML19081A001).

9. Letter from J. Holonich (NRC) to Brian Burgos (EPRI), “U.S. Nuclear Regulatory Commission Verification Letter for Electric Power Research Institute Topical Report MRP 227, Revision 1, ‘Materials Reliability Program: Pressurized Water Reactor Internals Inspection and Evaluations Guideline,’” February 19, 2020 (ADAMS Accession No. ML20006D152).

10. Email from J. Holonich (NRC) to K. Amberge (EPRI), "Transmittal of MRP-227, Rev 1-A Supplemental Information -A Verification," July 7, 2020 (ADAMS Accession No. ML20175A149).

11. NRC, "Revision 1 to the Final Safety Evaluation of Electric Power Research Institute (EPRI) Report, Materials Reliability Program (MRP) Report 1016596 (MRP-227), Revision 0, Pressurized Water Reactor (PWR) Internals Inspection and Evaluation Guidelines," December 16, 2011 (ADAMS Accession No. ML11308A770).

12. NRC Regulatory Information Summary 2011-07, "License Renewal Submittal Information for Pressurized Water Reactor Internals Aging Management," July 21, 2011 (ADAMS Accession No. ML111990086).

13. EPRI Technical Report No. 1022863, "Materials Reliability Program: Pressurized Water Reactor Internals Inspection and Evaluation Guidelines (MRP-227-A)," December 2011 (ADAMS Accession Nos. ML12017A194, ML12017A196, ML12017A197, ML12017A191, ML12017A192, ML12017A195, and ML12017A199).

14. EPRI Technical Report No. 1016596, "Materials Reliability Program: Pressurized Water Reactor Internals Inspection and Evaluation Guidelines (MRP-227 Revision 0)," December 2008 (ADAMS Accession Nos. ML090160204 (Cover letter from EPRI MRP) and ML090160206 (Final Report).

15. *U.S. Code of Federal Regulations*, "Domestic licensing of production and utilization facilities," Part 50, Chapter 1, Title 10, "Energy."

16. NEI 03-08, Revision 3, "Guideline for the Management of Materials Issues" February 2017 (ADAMS Accession No. ML19079A253).

# APPENDIX A

## PROPOSED REVISIONS TO SRP-SLR TABLE 3.1-1, "Summary of Aging Management Programs for Reactor Vessel, Internals, and Reactor Coolant System Evaluated in Chapter IV of the GALL-SLR Report"

Proposed revisions to NUREG-2192, "Standard Review Plan for Review of Subsequent License Renewal Applications for Nuclear Power Plants" (SRP-SLR), Table 3.1-1, "Summary of Aging Management Programs for Reactor Vessel, Internals, and Reactor Coolant System Evaluated in Chapter IV of the GALL-SLR Report," are provided in redline format. The revised items below supersede the respective items in SRP-SLR, Revision 0, Table 3.1-1.

**Table 3.1-1 Summary of Aging Management Programs for Reactor Vessel, Internals, and Reactor Coolant System Evaluated in Chapter IV of the GALL SLR Report**

| New, Modified, Deleted, Edited Item | ID | Type | Component | Aging Effect/Mechanism | Aging Management Program (AMP)/TLAA | Further Evaluation Recommended | GALL-SLR Item |
|---|---|---|---|---|---|---|---|
| M | 028 | PWR | Westinghouse-specific "Existing Programs" components: Stainless steel, nickel alloy ~~Westinghouse~~, and X-750 control rod guide tube support pins (split pins)~~, and Combustion Engineering thermal shield positioning pins; Zircaloy-4 Combustion Engineering incore instrumentation thimble tubes~~ exposed to reactor coolant and neutron flux | Loss of material due to wear; cracking due to SCC, ~~irradiation-assisted SCC~~IASCC, fatigue | AMP XI.M16A, "PWR Vessel Internals," and AMP XI.M2, "Water Chemistry" (for SCC mechanisms only) | Yes (SRP-SLR Section 3.1.2.2.9) | ~~IV.B2.RP-356~~<br>~~IV.B3.RP-357~~<br>~~IV.B3.RP-400~~<br>IV.B2.RP-355 (if AMP XI.M16A is credited for aging management)<br><br>IV.E.R-444 (if components are defined as ASME Section XI category components and the XI.M1 ISI AMP is credited for aging management)<br><br>IV.B2.RP-265 (if components can be placed in the "No Additional Measures" category) |
| M | 029 | BWR | Nickel alloy core shroud and core plate access hole cover (welded covers) exposed to reactor coolant | Cracking due to SCC, IGSCC, ~~irradiation-assisted SCC~~IASCC | AMP XI.M9, "BWR Vessel Internals," and AMP XI.M2, "Water Chemistry" | Yes (SRP-SLR Section 3.1.2.2.12) | IV.B1.R-94 |

**Table 3.1-1 Summary of Aging Management Programs for Reactor Vessel, Internals, and Reactor Coolant System Evaluated in Chapter IV of the GALL SLR Report**

| New, Modified, Deleted, Edited Item | ID | Type | Component | Aging Effect/Mechanism | Aging Management Program (AMP)/TLAA | Further Evaluation Recommended | GALL-SLR Item |
|---|---|---|---|---|---|---|---|
| ~~M~~D | 032 | ~~PWR~~ | ~~Stainless steel, nickel alloy, or CASS reactor vessel internals, core support structure (not already referenced as ASME Section XI Examination Category B-N-3 core support structure components in MRP-227-A), exposed to reactor coolant and neutron flux~~ | ~~Cracking, loss of material due to wear~~ | ~~AMP XI.M1, "ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD"~~ | ~~No~~ | ~~IV.B2.RP-382~~<br>~~IV.B3.RP-382~~<br>~~IV.B4.RP-382~~ |
| M | 041 | BWR | Nickel alloy core shroud and core plate access hole cover (mechanical covers) exposed to reactor coolant | Cracking due to SCC, IGSCC, ~~irradiation-assisted SCC~~IASCC | AMP XI.M9, "BWR Vessel Internals," and AMP XI.M2, "Water Chemistry" | Yes (SRP-SLR Section 3.1.2.2.12) | IV.B1.R-95 |
| M | 051a | PWR | Stainless steel, nickel alloy Babcock & Wilcox reactor internal "Primary" components exposed to reactor coolant, neutron flux | Cracking due to SCC, ~~irradiation-assisted SCC~~IASCC, fatigue | AMP XI.M16A, "PWR Vessel Internals," and AMP XI.M2, "Water Chemistry" (for SCC mechanisms only) | Yes (SRP-SLR Section 3.1.2.2.9) | IV.B4.RP-241<br>IV.B4.RP-241a<br>IV.B4.RP-242a<br>IV.B4.RP-247<br>IV.B4.RP-247a<br>IV.B4.RP-248<br>IV.B4.RP-248a<br>IV.B4.RP-249a<br>IV.B4.RP-252a<br>IV.B4.RP-252e<br>IV.B4.RP-256<br>IV.B4.RP-256a<br>IV.B4.RP-258a<br>IV.B4.RP-259a<br>IV.B4.RP-261<br>~~IV.B4.RP-400~~ |

**Table 3.1-1 Summary of Aging Management Programs for Reactor Vessel, Internals, and Reactor Coolant System Evaluated in Chapter IV of the GALL SLR Report**

| New, Modified, Deleted, Edited Item | ID | Type | Component | Aging Effect/Mechanism | Aging Management Program (AMP)/TLAA | Further Evaluation Recommended | GALL-SLR Item |
|---|---|---|---|---|---|---|---|
| M | 051b | PWR | Stainless steel, nickel alloy Babcock & Wilcox reactor internal "Expansion" components exposed to reactor coolant, neutron flux | Cracking due to SCC, ~~irradiation-assisted SCC~~IASCC, fatigue, overload | AMP XI.M16A, "PWR Vessel Internals," and AMP XI.M2, "Water Chemistry" (for SCC mechanisms only) | Yes (SRP-SLR Section 3.1.2.2.9) | IV.B4.RP-244<br>IV.B4.RP-244a<br>IV.B4.RP-245<br>IV.B4.RP-245a<br>IV.B4.RP-246<br>IV.B4.RP-246a<br>IV.B4.RP-246c<br>IV.B4.RP-246d<br>IV.B4.RP-252b<br>~~IV.B4.RP-254~~<br>~~IV.B4.RP-254a~~<br>IV.B4.RP-260a<br>IV.B4.RP-262<br>IV.B4.RP-352<br>IV.B4.RP-250a<br>IV.B4.RP-386 |
| M | 052a | PWR | Stainless steel, nickel alloy Combustion Engineering reactor internal "Primary" components exposed to reactor coolant, neutron flux | Cracking due to SCC, ~~irradiation-assisted SCC~~IASCC, fatigue | AMP XI.M16A, "PWR Vessel Internals," and AMP XI.M2, "Water Chemistry" (for SCC mechanisms only) | Yes (SRP-SLR Section 3.1.2.2.9) | IV.B3.RP-312<br>IV.B3.RP-314<br>IV.B3.RP-322<br>IV.B3.RP-324<br>IV.B3.RP-326a<br>IV.B3.RP-327<br>IV.B3.RP-328<br>IV.B3.RP-342<br>IV.B3.RP-358<br>IV.B3.RP-362a<br>~~IV.B3.RP-363~~<br>IV.B3.RP-338<br>IV.B3.RP-343 |

**Table 3.1-1 Summary of Aging Management Programs for Reactor Vessel, Internals, and Reactor Coolant System Evaluated in Chapter IV of the GALL SLR Report**

| New, Modified, Deleted, Edited Item | ID | Type | Component | Aging Effect/Mechanism | Aging Management Program (AMP)/TLAA | Further Evaluation Recommended | GALL-SLR Item |
|---|---|---|---|---|---|---|---|
| M | 052b | PWR | Stainless steel, nickel alloy Combustion Engineering reactor internal "Expansion" components exposed to reactor coolant, neutron flux | Cracking due to SCC, ~~irradiation-assisted SCC~~IASCC, fatigue | AMP XI.M16A, "PWR Vessel Internals," and AMP XI.M2, "Water Chemistry" (for SCC mechanisms only) | Yes (SRP-SLR Section 3.1.2.2.9) | IV.B3.RP-313<br>IV.B3.RP-316<br>IV.B3.RP-323<br>IV.B3.RP-325<br>IV.B3.RP-329<br>IV.B3.RP-330<br>IV.B3.RP-333<br>IV.B3.RP-335<br>IV.B3.RP-362c<br>IV.B3.RP-363 |
| M | 052c | PWR | Stainless steel, nickel alloy Combustion Engineering reactor internal "Existing Programs" components exposed to reactor coolant, neutron flux | Cracking due to SCC, ~~irradiation-assisted SCC~~IASCC, fatigue | AMP XI.M16A, "PWR Vessel Internals," and AMP XI.M2, "Water Chemistry" (for SCC mechanisms only) | Yes (SRP-SLR Section 3.1.2.2.9) | IV.B3.RP-320<br>IV.B3.RP-320a<br>IV.B3.RP-334 |
| M | 053a | PWR | Stainless steel, nickel alloy Westinghouse reactor internal "Primary" components exposed to reactor coolant, neutron flux | Cracking due to SCC, ~~irradiation-assisted SCC~~IASCC, fatigue | AMP XI.M16A, "PWR Vessel Internals," and AMP XI.M2, "Water Chemistry" (for SCC mechanisms only) | Yes (SRP-SLR Section 3.1.2.2.9) | IV.B2.RP-270a<br>IV.B2.RP-271<br>IV.B2.RP-275<br>IV.B2.RP-276<br>~~IV.B2.RP-280~~<br>IV.B2.RP-296a<br>IV.B2.RP-298<br>IV.B2.RP-302<br>IV.B2.RP-387 |
| M | 053b | PWR | Stainless steel Westinghouse reactor internal "Expansion" components exposed to reactor coolant and neutron flux | Cracking due to SCC, ~~irradiation-assisted SCC~~IASCC, fatigue | AMP XI.M16A, "PWR Vessel Internals," and AMP XI.M2, "Water Chemistry" (for SCC mechanisms only) | Yes (SRP-SLR Section 3.1.2.2.9) | IV.B2.RP-273<br>~~IV.B2.RP-278~~<br>IV.B2.RP-280<br>IV.B2.RP-286<br>IV.B2.RP-291<br>IV.B2.RP-291a<br>IV.B2.RP-291b<br>IV.B2.RP-293<br>IV.B2.RP-294<br>IV.B2.RP-298a<br>IV.B2.RP-387a |

**Table 3.1-1 Summary of Aging Management Programs for Reactor Vessel, Internals, and Reactor Coolant System Evaluated in Chapter IV of the GALL SLR Report**

| New, Modified, Deleted, Edited Item | ID | Type | Component | Aging Effect/Mechanism | Aging Management Program (AMP)/TLAA | Further Evaluation Recommended | GALL-SLR Item |
|---|---|---|---|---|---|---|---|
| M | 053c | PWR | Stainless steel, nickel alloy, or stellite Westinghouse reactor internal "Existing Programs" components exposed to reactor coolant, neutron flux | Cracking due to SCC, ~~irradiation-assisted SCC~~IASCC, fatigue | AMP XI.M16A, "PWR Vessel Internals," and AMP XI.M2, "Water Chemistry" (for SCC mechanisms only) | Yes (SRP-SLR Section 3.1.2.2.9) | IV.B2.RP-289<br>IV.B2.RP-301<br>IV.B2.RP-345a<br>IV.B2.RP-346<br>IV.B2.RP-399<br>~~IV.B2.RP-355~~ |
| M | 054 | PWR | Stainless steel Westinghouse-design bottom mounted instrument system flux thimble tubes (with or without chrome plating) exposed to reactor coolant and neutron flux | Loss of material due to wear | AMP XI.M37, "Flux Thimble Tube Inspection" | No | IV.B2.RP-284 |
| M | 056a | PWR | Stainless steel (SS, including CASS, PH SS or martensitic SS) or nickel alloy Combustion Engineering reactor internal "Primary" components exposed to reactor coolant and neutron flux | Loss of fracture toughness due to neutron irradiation embrittlement and for CASS, martensitic SS, and PH SS due to thermal aging embrittlement; changes in dimensions due to void swelling, distortion; loss of preload due to thermal and irradiation-enhanced stress relaxation, creep; loss of material due to wear | AMP XI.M16A, "PWR Vessel Internals" | Yes (SRP-SLR Section 3.1.2.2.9) | IV.B3.RP-315<br>IV.B3.RP-318<br>IV.B3.RP-359<br>IV.B3.RP-360<br>IV.B3.RP-362<br>~~IV.B3.RP-364~~<br>IV.B3.RP-366<br>IV.B3.RP-365<br>IV.B3.RP-326<br>IV.B3.RP-338a |

**Table 3.1-1 Summary of Aging Management Programs for Reactor Vessel, Internals, and Reactor Coolant System Evaluated in Chapter IV of the GALL SLR Report**

| New, Modified, Deleted, Edited Item | ID | Type | Component | Aging Effect/Mechanism | Aging Management Program (AMP)/TLAA | Further Evaluation Recommended | GALL-SLR Item |
|---|---|---|---|---|---|---|---|
| M | 056b | PWR | Stainless steel (SS, including CASS, PH SS or martensitic SS) Combustion Engineering "Expansion" reactor internal components exposed to reactor coolant and neutron flux | Loss of fracture toughness due to neutron irradiation embrittlement and for CASS, martensitic SS, and PH SS due to thermal aging embrittlement; changes in dimensions due to void swelling, distortion; loss of preload due to thermal and irradiation-enhanced stress relaxation, creep; loss of material due to wear | AMP XI.M16A, "PWR Vessel Internals" | Yes (SRP-SLR Section 3.1.2.2.9) | IV.B3.RP-317<br>IV.B3.RP-331<br>IV.B3.RP-333a<br>IV.B3.RP-359a<br>IV.B3.RP-361<br>IV.B3.RP-362b<br>IV.B3.R-455<br>IV.B3.RP-364 |
| M | 056c | PWR | Stainless steel (SS, including CASS, PH SS or martensitic SS) or nickel alloy Combustion Engineering reactor internal "Existing Programs" components exposed to reactor coolant and neutron flux | Loss of fracture toughness due to neutron irradiation embrittlement and for CASS, martensitic SS, and PH SS due to thermal aging embrittlement; changes in dimensions due to void swelling, distortion; loss of preload due to thermal and irradiation-enhanced stress relaxation, creep; loss of material due to wear | AMP XI.M16A, "PWR Vessel Internals" | Yes (SRP-SLR Section 3.1.2.2.9) | IV.B3.RP-319<br>IV.B3.RP-332<br>~~IV.B3.RP-334a~~<br>IV.B3.RP-336<br>IV.B3.RP-357 |

**Table 3.1-1 Summary of Aging Management Programs for Reactor Vessel, Internals, and Reactor Coolant System Evaluated in Chapter IV of the GALL SLR Report**

| New, Modified, Deleted, Edited Item | ID | Type | Component | Aging Effect/Mechanism | Aging Management Program (AMP)/TLAA | Further Evaluation Recommended | GALL-SLR Item |
|---|---|---|---|---|---|---|---|
| M | 058a | PWR | Stainless steel (SS, including CASS, PH SS or martensitic SS), nickel alloy Babcock & Wilcox reactor internal "Primary" components exposed to reactor coolant and neutron flux | Loss of fracture toughness due to neutron irradiation embrittlement and for CASS, martensitic SS, and PH SS due to thermal aging embrittlement; or changes in dimensions due to void swelling or distortion; or loss of preload due to wear; or loss of material due to wear | AMP XI.M16A, "PWR Vessel Internals" | Yes (SRP-SLR Section 3.1.2.2.9) | IV.B4.RP-240<br>IV.B4.RP-240a<br>IV.B4.RP-242<br>IV.B4.RP-247b<br>IV.B4.RP-247c<br>IV.B4.RP-248b<br>IV.B4.RP-249<br>IV.B4.RP-251<br>IV.B4.RP-251a<br>IV.B4.RP-252<br>IV.B4.RP-252d<br>IV.B4.RP-256b<br>IV.B4.RP-258<br>IV.B4.RP-259<br>~~IV.B4.RP-401~~ |
| M | 058b | PWR | Stainless steel (SS, including CASS, PH SS or martensitic SS), nickel alloy Babcock & Wilcox reactor internal "Expansion" components exposed to reactor coolant and neutron flux | Loss of fracture toughness due to neutron irradiation embrittlement and for CASS, martensitic SS, and PH SS due to thermal aging embrittlement; or changes in dimensions due to void swelling, or distortion; or loss of preload due to thermal and irradiation-enhanced stress relaxation, or creep; or loss of material due to wear | AMP XI.M16A, "PWR Vessel Internals" | Yes (SRP-SLR Section 3.1.2.2.9) | IV.B4.RP-245b<br>IV.B4.RP-245c<br>IV.B4.RP-246b<br>IV.B4.RP-246e<br>~~IV.B4.RP-254b~~<br>IV.B4.RP-260<br>IV.B4.RP-243<br>IV.B4.RP-243a<br>IV.B4.RP-250<br>IV.B4.RP-252c<br>IV.B4.RP-386a |

**Table 3.1-1 Summary of Aging Management Programs for Reactor Vessel, Internals, and Reactor Coolant System Evaluated in Chapter IV of the GALL SLR Report**

| New, Modified, Deleted, Edited Item | ID | Type | Component | Aging Effect/Mechanism | Aging Management Program (AMP)/TLAA | Further Evaluation Recommended | GALL-SLR Item |
|---|---|---|---|---|---|---|---|
| M | 059b | PWR | Stainless steel (SS, including CASS, PH SS or martensitic SS) Westinghouse reactor internal "Expansion" components exposed to reactor coolant and neutron flux | Loss of fracture toughness due to neutron irradiation embrittlement and for CASS, martensitic SS, and PH SS due to thermal aging embrittlement; changes in dimensions due to void swelling, distortion; loss of preload due to thermal and irradiation-enhanced stress relaxation, creep; loss of material due to wear | AMP XI.M16A, "PWR Vessel Internals" | Yes (SRP-SLR Section 3.1.2.2.9) | IV.B2.RP-274<br>~~IV.B2.RP-278a~~<br>IV.B2.RP-280a<br>IV.B2.RP-287<br>IV.B2.RP-290<br>IV.B2.RP-290a<br>IV.B2.RP-290b<br>IV.B2.RP-292<br>IV.B2.RP-295<br>IV.B2.RP-297a<br>IV.B2.RP-388a |
| M | 059c | PWR | Stainless steel (SS, including CASS, PH SS or martensitic SS), ~~or~~ nickel alloy, or stellite Westinghouse reactor internal "Existing Programs" components exposed to reactor coolant and neutron flux | Loss of fracture toughness due to neutron irradiation embrittlement and for CASS, martensitic SS, and PH SS due to thermal aging embrittlement; changes in dimensions due to void swelling, distortion; loss of preload due to thermal and irradiation-enhanced stress relaxation, creep; loss of material due to wear | AMP XI.M16A, "PWR Vessel Internals" | Yes (SRP-SLR Section 3.1.2.2.9) | IV.B2.RP-285<br>IV.B2.RP-288<br>IV.B2.RP-299<br>IV.B2.RP-345 |
| M | 103 | BWR | Stainless steel, nickel alloy reactor internal components exposed to reactor coolant and neutron flux | Cracking due to SCC, IGSCC, ~~irradiation-assisted SCC~~IASCC | AMP XI.M9, "BWR Vessel Internals," and AMP XI.M2, "Water Chemistry" | Yes (SRP-SLR Section 3.1.2.2.12) | IV.B1.R-422<br>IV.B1.R-100<br>IV.B1.R-105<br>IV.B1.R-92<br>IV.B1.R-93<br>IV.B1.R-96<br>IV.B1.R-97<br>IV.B1.R-98<br>IV.B1.R-99 |

**Table 3.1-1 Summary of Aging Management Programs for Reactor Vessel, Internals, and Reactor Coolant System Evaluated in Chapter IV of the GALL SLR Report**

| New, Modified, Deleted, Edited Item | ID | Type | Component | Aging Effect/Mechanism | Aging Management Program (AMP)/TLAA | Further Evaluation Recommended | GALL-SLR Item |
|---|---|---|---|---|---|---|---|
| N | 114 | BWR/PWR | Reactor coolant system components defined as ASME Section XI Code Class components (ASME Code Class 1 reactor coolant pressure boundary components, reactor vessel interior attachments, or core support structure components~~,~~; or ASME Class 2 or 3 components - including ASME defined appurtenances, component supports, and associated pressure boundary welds, or components subject to plant-specific equivalent classifications for these ASME code classes) | Cracking due to SCC, IGSCC, PWSCC, IASCC (SCC mechanisms for stainless steel, nickel alloy components only), fatigue, or cyclic loading; loss of material due to general corrosion (steel only), pitting corrosion, crevice corrosion, or wear | AMP XI.M1, "ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD," and AMP XI.M2, "Water Chemistry" (water chemistry-related or corrosion-related aging effect mechanisms only) | No | IV.E.R-444 |
| N | 118 | PWR | Stainless steel, nickel alloy PWR reactor vessel internal components exposed to reactor coolant, neutron flux | Cracking due to SCC, ~~irradiation-assisted SCC~~IASCC, cyclic loading, fatigue | Plant-specific aging management program or AMP XI.M16A, "PWR Vessel Internals," and AMP XI.M2, "Water Chemistry" (SCC and IASCC only), with adjusted site-specific or component-specific aging management basis for a given component | Yes (SRP-SLR Section 3.1.2.2.9) | IV.B2.R-423<br>IV.B3.R-423<br>IV.B4.R-423 |

| Table 3.1-1 Summary of Aging Management Programs for Reactor Vessel, Internals, and Reactor Coolant System Evaluated in Chapter IV of the GALL SLR Report | | | | | | | |
|---|---|---|---|---|---|---|---|
| **New, Modified, Deleted, Edited Item** | **ID** | **Type** | **Component** | **Aging Effect/Mechanism** | **Aging Management Program (AMP)/TLAA** | **Further Evaluation Recommended** | **GALL-SLR Item** |
| N | 119 | PWR | Stainless steel, nickel alloy PWR reactor vessel internal components exposed to reactor coolant, neutron flux | Loss of fracture toughness due to neutron irradiation embrittlement or thermal aging embrittlement; changes in dimensions due to void swelling or distortion; loss of preload due to thermal and irradiation-enhanced stress relaxation or creep; loss of material due to wear | Plant-specific aging management program or AMP XI.M16A, "PWR Vessel Internals," with adjusted site-specific or component-specific aging management basis for a given component | Yes (SRP-SLR Section 3.1.2.2.9) | IV.B2.R-424<br>IV.B3.R-424<br>IV.B4.R-424 |

# APPENDIX B

# PROPOSED REVISIONS TO GALL-SLR REPORT TABLES IV.B2, IV.B3, AND IV.B4

## APPENDIX B.1

## PROPOSED REVISIONS TO GALL-SLR REPORT TABLE IV.B2, "REACTOR VESSEL INTERNALS (PWR)—WESTINGHOUSE"

NUREG-2191, Volumes 1 and 2, "Generic Aging Lessons Learned for Subsequent License Renewal (GALL-SLR) Report," Table IV.B2, "Reactor Vessel Internals (PWR)—Westinghouse," addresses the Westinghouse pressurized-water reactor (PWR) vessel internals, which consist of components in the upper internals assembly, the control rod guide tube (CRGT) assembly, the core barrel assembly, the baffle/former assembly, the lower internals assembly, lower support assembly, thermal shield assembly, bottom-mounted instrumentation system, and alignment and interfacing components.

Proposed revisions to Table IV.B2 of the GALL-SLR Report are provided in redline format. These AMR items supersede the respective items in GALL-SLR Report, Revision 0, Table IV.B2.

GALL-SLR Report Table IV.B2 Proposed Revisions

| IV Table B2 | REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM Reactor Vessel Internals (PWR)—Westinghouse | | | | | | | |
|---|---|---|---|---|---|---|---|---|
| **New, Modified, Deleted, Edited Item** | **Item** | **SRP Item (Table, ID)** | **Structure and/or Component** | **Material** | **Environment** | **Aging Effect/Mechanism** | **Aging Management Program (AMP)/TLAA** | **Further Evaluation** |
| M | IV.B2.RP-301 | 3.1-1, 053c | Alignment and interfacing components: upper core plate alignment pins (fuel alignment pins) | Stainless steel | Reactor coolant and neutron flux | Cracking due to ~~SCC~~IASCC or fatigue | AMP XI.M16A, "PWR Vessel Internals," and AMP XI.M2, "Water Chemistry" (for SCC mechanisms only) | Yes |
| M | IV.B2.RP-299 | 3.1-1, 059c | Alignment and interfacing components: upper core plate alignment pins (fuel alignment pins) | Stainless steel | Reactor coolant and neutron flux | Loss of material due to wear; loss of fracture toughness due to neutron irradiation embrittlement | AMP XI.M16A, "PWR Vessel Internals" | Yes |
| M | IV.B2.RP-271 | 3.1-1, 053a | Baffle-to-former assembly: ~~accessible~~ baffle-to-former bolts | Stainless steel | Reactor coolant and neutron flux | Cracking due to ~~irradiation-assisted SCC~~IASCC or fatigue | AMP XI.M16A, "PWR Vessel Internals," and AMP XI.M2, "Water Chemistry" (for SCC mechanisms only) | Yes |

| IV Table B2 | REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM Reactor Vessel Internals (PWR)—Westinghouse | | | | | | | |
|---|---|---|---|---|---|---|---|---|
| **New, Modified, Deleted, Edited Item** | **Item** | **SRP Item (Table, ID)** | **Structure and/or Component** | **Material** | **Environment** | **Aging Effect/Mechanism** | **Aging Management Program (AMP)/TLAA** | **Further Evaluation** |
| M | IV.B2.RP-272 | 3.1-1, 059a | Baffle-to-former assembly: ~~accessible~~ baffle-to-former bolts | Stainless steel | Reactor coolant and neutron flux | Loss of fracture toughness due to neutron irradiation embrittlement; changes in dimensions due to void swelling or distortion; loss of preload due to thermal and irradiation-enhanced stress relaxation or creep; loss of material due to wear | AMP XI.M16A, "PWR Vessel Internals" | Yes |
| M | IV.B2.RP-270 | 3.1-1, 059a | Baffle-to-former assembly: baffle and former plates | Stainless steel | Reactor coolant and neutron flux | Changes in dimensions due to void swelling or distortion; loss of fracture toughness due to neutron irradiation embrittlement | AMP XI.M16A, "PWR Vessel Internals" | Yes |
| M | IV.B2.RP-270a | 3.1-1, 053a | Baffle-to-former assembly: baffle and former plates | Stainless steel | Reactor coolant and neutron flux | Cracking due to ~~irradiation-assisted SCC~~IASCC or fatigue | AMP XI.M16A, "PWR Vessel Internals," and AMP XI.M2, "Water Chemistry" (for SCC mechanisms only) | Yes |

| IV<br>Table B2 | REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM<br>Reactor Vessel Internals (PWR)—Westinghouse | | | | | | | |
|---|---|---|---|---|---|---|---|---|
| **New, Modified, Deleted, Edited Item** | **Item** | **SRP Item (Table, ID)** | **Structure and/or Component** | **Material** | **Environment** | **Aging Effect/Mechanism** | **Aging Management Program (AMP)/TLAA** | **Further Evaluation** |
| M | IV.B2.RP-275 | 3.1-1, 053a | Baffle-to-former assembly: baffle-edge bolts ~~(all plants with baffle-edge~~, corner bolts~~)~~ | Stainless steel | Reactor coolant and neutron flux | Cracking due to ~~irradiation-assisted SCC~~IASCC or fatigue | AMP XI.M16A, "PWR Vessel Internals," and AMP XI.M2, "Water Chemistry" (for SCC mechanisms only) | Yes |
| M | IV.B2.RP-354 | 3.1-1, 059a | Baffle-to-former assembly: baffle-edge bolts ~~(all plants with baffle-edge~~, corner bolts~~)~~ | Stainless steel | Reactor coolant and neutron flux | Loss of fracture toughness due to neutron irradiation embrittlement; changes in dimensions due to void swelling or distortion; loss of preload due to thermal and irradiation-enhanced stress relaxation or creep; loss of material due to wear | AMP XI.M16A, "PWR Vessel Internals" | Yes |
| M | IV.B2.RP-273 | 3.1-1, 053b | Baffle-to-former assembly: barrel-to-former bolts | Stainless steel | Reactor coolant and neutron flux | Cracking due to ~~irradiation-assisted SCC~~IASCC or fatigue | AMP XI.M16A, "PWR Vessel Internals," and AMP XI.M2, "Water Chemistry" (for SCC mechanisms only) | Yes |

| IV<br>Table B2 | REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM<br>Reactor Vessel Internals (PWR)—Westinghouse | | | | | | | |
|---|---|---|---|---|---|---|---|---|
| **New, Modified, Deleted, Edited Item** | **Item** | **SRP Item (Table, ID)** | **Structure and/or Component** | **Material** | **Environment** | **Aging Effect/Mechanism** | **Aging Management Program (AMP)/TLAA** | **Further Evaluation** |
| M | IV.B2.RP-274 | 3.1-1, 059b | Baffle-to-former assembly: barrel-to-former bolts | Stainless steel | Reactor coolant and neutron flux | Loss of fracture toughness due to neutron irradiation embrittlement; changes in dimensions due to void swelling or distortion; loss of preload due to thermal and irradiation-enhanced stress relaxation or creep; loss of material due to wear | AMP XI.M16A, "PWR Vessel Internals" | Yes |
| M | IV.B2.RP-293 | 3.1-1, 053b | Bottom-mounted instrumentation system: bottom-mounted instrumentation (BMI) column bodies | Stainless steel | Reactor coolant and neutron flux | Cracking due to SCC, IASCC, or fatigue | AMP XI.M16A, "PWR Vessel Internals," and AMP XI.M2, "Water Chemistry" (for SCC mechanisms only) | Yes |
| M | IV.B2.RP-292 | 3.1-1, 059b | Bottom-mounted instrumentation system: bottom-mounted instrumentation (BMI) column bodies | Stainless steel | Reactor coolant and neutron flux | Loss of fracture toughness due to neutron irradiation embrittlement; changes in dimension due to void swelling or distortion; loss of material due to wear | AMP XI.M16A, "PWR Vessel Internals" | Yes |

| IV<br>Table B2 | REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM<br>Reactor Vessel Internals (PWR)—Westinghouse | | | | | | | |
|---|---|---|---|---|---|---|---|---|
| **New, Modified, Deleted, Edited Item** | **Item** | **SRP Item (Table, ID)** | **Structure and/or Component** | **Material** | **Environment** | **Aging Effect/Mechanism** | **Aging Management Program (AMP)/TLAA** | **Further Evaluation** |
| M | IV.B2.RP-296 | 3.1-1, 059a | Control rod guide tube (CRGT) assemblies: CRGT guide plates (cards) | Stainless steel (including CASS) | Reactor coolant and neutron flux | Loss of material due to wear; loss of fracture due to neutron irradiation embrittlement, and for CASS, thermal aging embrittlement) | AMP XI.M16A, "PWR Vessel Internals" | Yes |
| N | IV.B2.RP-296a | 3.1-1, 053a | Control rod guide tube (CRGT) assemblies: CRGT guide plates (cards) | Stainless steel (including CASS) | Reactor coolant and neutron flux | Cracking due to SCC or fatigue | AMP XI.M16A, "PWR Vessel Internals," and AMP XI.M2, "Water Chemistry" (for SCC mechanisms only) | Yes |
| M | IV.B2.RP-298 | 3.1-1, 053a | Control rod guide tube (CRGT) assemblies: ~~CRGT~~ lower flange welds ~~(accessible)~~in outer (peripheral) CRGT assemblies | Stainless steel | Reactor coolant and neutron flux | Cracking due to SCC, IASCC, or fatigue | AMP XI.M16A, "PWR Vessel Internals," and AMP XI.M2, "Water Chemistry" (for SCC mechanisms only) | Yes |
| N | IV.B2.RP-298a | 3.1-1, 053b | Control rod guide tube (CRGT) assemblies: lower flange welds in remaining (non-peripheral) CRGT assemblies | Stainless steel | Reactor coolant and neutron flux | Cracking due to SCC, IASCC, or fatigue | AMP XI.M16A, "PWR Vessel Internals," and AMP XI.M2, "Water Chemistry" (for SCC mechanisms only) | Yes |

| IV Table B2 | REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM Reactor Vessel Internals (PWR)—Westinghouse | | | | | | | |
|---|---|---|---|---|---|---|---|---|
| **New, Modified, Deleted, Edited Item** | **Item** | **SRP Item (Table, ID)** | **Structure and/or Component** | **Material** | **Environment** | **Aging Effect/Mechanism** | **Aging Management Program (AMP)/TLAA** | **Further Evaluation** |
| M | IV.B2.RP-297 | 3.1-1, 059a | Control rod guide tube (CRGT) assemblies: ~~CRGT~~ lower flange welds ~~(accessible)~~in outer (peripheral) CRGT assemblies | Stainless steel (including CASS) | Reactor coolant and neutron flux | Loss of fracture toughness due to thermal aging and neutron irradiation embrittlement and for CASS, due to thermal aging embrittlement | AMP XI.M16A, "PWR Vessel Internals" | Yes |
| N | IV.B2.RP-297a | 3.1-1, 059b | Control rod guide tube (CRGT) assemblies: lower flange welds in the remaining (non-peripheral) CRGT assemblies | Stainless steel (including CASS) | Reactor coolant and neutron flux | Loss of fracture toughness due to neutron irradiation embrittlement, and for CASS, due to thermal aging embrittlement | AMP XI.M16A, "PWR Vessel Internals" | Yes |
| M | IV.B2.RP-355 | 3.1-1, ~~053c~~028 | Control rod guide tube (CRGT) assemblies: guide tube support pins (split pins) | ~~Stainless steel, nickel~~Nickel alloy (X-750) | Reactor coolant and neutron flux | Cracking due to SCC or fatigue; loss of material due to wear | AMP XI.M16A, "PWR Vessel Internals," and AMP XI.M2, "Water Chemistry" (for SCC mechanisms only) – using component-specific evaluation per MRP guidelines | Yes |
| ~~M~~D | IV.B2.RP-356 | ~~3.1-1, 028~~ | ~~Control rod guide tube (CRGT) assemblies: guide tube support pins (split pins)~~ | ~~Stainless steel, nickel alloy~~ | ~~Reactor coolant and neutron flux~~ | ~~Loss of material due to wear~~ | ~~AMP XI.M16A, "PWR Vessel Internals"~~ | ~~Yes~~ |

| IV Table B2 | REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM Reactor Vessel Internals (PWR)—Westinghouse | | | | | | | |
|---|---|---|---|---|---|---|---|---|
| **New, Modified, Deleted, Edited Item** | **Item** | **SRP Item (Table, ID)** | **Structure and/or Component** | **Material** | **Environment** | **Aging Effect/Mechanism** | **Aging Management Program (AMP)/TLAA** | **Further Evaluation** |
| M | IV.B2.RP-345 | 3.1-1, 059c | Core barrel assembly: core barrel flange | Stainless steel | Reactor coolant and neutron flux | Loss of material due to wear | AMP XI.M16A, "PWR Vessel Internals~~," and AMP XI.M2, "Water Chemistry" (for SCC mechanisms only)"~~ | Yes |
| N | IV.B2.RP-345a | 3.1-1, 053c | Core barrel assembly: core barrel flange | Stainless steel | Reactor coolant and neutron flux | Cracking due to SCC or fatigue | AMP XI.M16A, "PWR Vessel Internals," and AMP XI.M2, "Water Chemistry" (for SCC mechanisms only) | Yes |
| ~~M~~D | IV.B2.RP-278 | ~~3.1-1, 053b~~ | ~~Core barrel assembly: core barrel outlet nozzle welds~~ | ~~Stainless steel~~ | ~~Reactor coolant and neutron flux~~ | ~~Cracking due to SCC or fatigue~~ | ~~AMP XI.M16A, "PWR Vessel Internals," and AMP XI.M2, "Water Chemistry" (for SCC mechanisms only)~~ | ~~Yes~~ |
| ~~M~~D | IV.B2.RP-278a | ~~3.1-1, 059b~~ | ~~Core barrel assembly: core barrel outlet nozzle welds~~ | ~~Stainless steel~~ | ~~Reactor coolant and neutron flux~~ | ~~Loss of fracture toughness due to neutron irradiation embrittlement~~ | ~~AMP XI.M16A, "PWR Vessel Internals"~~ | ~~Yes~~ |

| IV<br>Table B2 | REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM<br>Reactor Vessel Internals (PWR)—Westinghouse | | | | | | | |
|---|---|---|---|---|---|---|---|---|
| **New, Modified, Deleted, Edited Item** | **Item** | **SRP Item (Table, ID)** | **Structure and/or Component** | **Material** | **Environment** | **Aging Effect/Mechanism** | **Aging Management Program (AMP)/TLAA** | **Further Evaluation** |
| M | IV.B2.RP-280 | 3.1-1, ~~053a~~053b | Core barrel assembly: lower flange weld (core barrel ~~flange weld~~-to-support plate weld), upper circumferential (girth) weld, and upper vertical (axial) welds | Stainless steel | Reactor coolant and neutron flux | Cracking due to SCC, ~~irradiation-assisted SCC,~~IASCC (lower flange weld only), or fatigue | AMP XI.M16A, "PWR Vessel Internals," and AMP XI.M2, "Water Chemistry" (for SCC mechanisms only) | Yes |
| N | IV.B2.RP-280a | 3.1-1, 059b | Core barrel assembly; lower flange weld (core barrel-to-support plate weld) | Stainless steel | Reactor coolant and neutron flux | Loss of fracture toughness due to neutron irradiation embrittlement | AMP XI.M16A, "PWR Vessel Internals" | Yes |
| M | IV.B2.RP-387 | 3.1-1, 053a | Core barrel assembly~~: upper core barrel and~~ lower ~~core barrel~~ circumferential (girth) welds | Stainless steel | Reactor coolant and neutron flux | Cracking due to SCC, ~~irradiation-assisted SCC~~IASCC, or fatigue | AMP XI.M16A, "PWR Vessel Internals," and AMP XI.M2, "Water Chemistry" (for SCC mechanisms only) | Yes |
| M | IV.B2.RP-388 | 3.1-1, 059a | Core barrel assembly: ~~upper core barrel and~~ lower ~~core barrel~~ circumferential (girth) welds | Stainless steel | Reactor coolant and neutron flux | Loss of fracture toughness due to neutron irradiation embrittlement, changes in dimension due to void swelling or distortion | AMP XI.M16A, "PWR Vessel Internals" | Yes |

| IV<br>Table B2 | REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM<br>Reactor Vessel Internals (PWR)—Westinghouse | | | | | | | |
|---|---|---|---|---|---|---|---|---|
| **New, Modified, Deleted, Edited Item** | **Item** | **SRP Item (Table, ID)** | **Structure and/or Component** | **Material** | **Environment** | **Aging Effect/Mechanism** | **Aging Management Program (AMP)/TLAA** | **Further Evaluation** |
| M | IV.B2.RP-387a | 3.1-1, 053b | Core barrel assembly: ~~upper core barrel and lower core barrel~~middle vertical (axial) welds and lower vertical (axial) welds | Stainless steel | Reactor coolant and neutron flux | Cracking due to SCC, ~~irradiation-assisted SCC~~IASCC, or fatigue | AMP XI.M16A, "PWR Vessel Internals," and AMP XI.M2, "Water Chemistry" (for SCC mechanisms only) | Yes |
| M | IV.B2.RP-388a | 3.1-1, 059b | Core barrel assembly: ~~upper core barrel and lower core barrel~~middle vertical (axial) welds and lower vertical (axial) welds | Stainless steel | Reactor coolant and neutron flux | Loss of fracture toughness due to neutron irradiation embrittlement; changes in dimension due to void swelling or distortion | AMP XI.M16A, "PWR Vessel Internals" | Yes |
| M | IV.B2.RP-276 | 3.1-1, 053a | Core barrel assembly: upper ~~core barrel~~ flange weld | Stainless steel | Reactor coolant and neutron flux | Cracking due to ~~irradiation-assisted~~ SCC or fatigue | AMP XI.M16A, "PWR Vessel Internals," and AMP XI.M2, "Water Chemistry" (for SCC mechanisms only) | Yes |
| M | IV.B2.RP-285 | 3.1-1, 059c | ~~Lower internals assembly~~Alignment and interfacing components: clevis ~~insert~~ inserts (including bolts or screws, and clevis insert surfaces) | ~~Nickel alloy~~Stainless steel, nickel alloy (including alloy 600, X-750), stellite (for insert surfaces only) | Reactor coolant and neutron flux | Loss of material due to wear; loss of preload due to thermal ~~and~~or irradiation-enhanced stress relaxation or creep (bolts and screws only); changes in dimension due to distortion | AMP XI.M16A, "PWR Vessel Internals" | Yes |

| **IV**<br>**Table B2** | **REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM**<br>**Reactor Vessel Internals (PWR)—Westinghouse** | | | | | | | |
|---|---|---|---|---|---|---|---|---|
| **New, Modified, Deleted, Edited Item** | **Item** | **SRP Item (Table, ID)** | **Structure and/or Component** | **Material** | **Environment** | **Aging Effect/Mechanism** | **Aging Management Program (AMP)/TLAA** | **Further Evaluation** |
| M | IV.B2.RP-399 | 3.1-1, 053c | ~~Lower internals assembly~~Alignment and interfacing components: clevis ~~insert~~ inserts (including bolts or screws~~,~~, dowels, and clevis insert surfaces) | Stainless steel, nickel alloy (including Alloy 600, X-750), stellite (for insert surfaces only) | Reactor coolant and neutron flux | Cracking due to ~~primary water~~ SCC~~,~~ ~~irradiation-assisted SCC,~~ or fatigue | AMP XI.M16A, "PWR Vessel Internals," and AMP XI.M2, "Water Chemistry" (for SCC mechanisms only) | Yes |
| M | IV.B2.RP-289 | 3.1-1, 053c | Lower internals assembly: lower core plate ~~and~~or extra-long (XL) lower core plate | Stainless steel | Reactor coolant and neutron flux | Cracking due to ~~irradiation-assisted SCC~~IASCC or fatigue | AMP XI.M16A, "PWR Vessel Internals," and AMP XI.M2, "Water Chemistry" (for SCC mechanisms only) | Yes |
| M | IV.B2.RP-288 | 3.1-1, 059c | Lower internals assembly: lower core plate ~~and~~or extra-long (XL) lower core plate | Stainless steel | Reactor coolant and neutron flux | Loss of fracture toughness due to neutron irradiation embrittlement; loss of material due to wear | AMP XI.M16A, "PWR Vessel Internals" | Yes |
| M | IV.B2.RP-291a | 3.1-1, 053b | Lower ~~support~~internals assembly: lower support forging or casting | Stainless steel | Reactor coolant and neutron flux | Cracking due to SCC or fatigue | AMP XI.M16A, "PWR Vessel Internals," and AMP XI.M2, "Water Chemistry" (for SCC mechanisms only) | Yes |

| IV<br>Table B2 | REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM<br>Reactor Vessel Internals (PWR)—Westinghouse | | | | | | | |
|---|---|---|---|---|---|---|---|---|
| **New, Modified, Deleted, Edited Item** | **Item** | **SRP Item (Table, ID)** | **Structure and/or Component** | **Material** | **Environment** | **Aging Effect/Mechanism** | **Aging Management Program (AMP)/TLAA** | **Further Evaluation** |
| M | IV.B2.RP-290a | 3.1-1, 059b | Lower ~~support~~internals assembly: lower support forging or casting | Stainless steel | Reactor coolant and neutron flux | Loss of fracture toughness due to neutron irradiation embrittlement (and thermal aging embrittlement for CASS, PH SS, and martensitic SS) | AMP XI.M16A, "PWR Vessel Internals" | Yes |
| M | IV.B2.RP-291 | 3.1-1, 053b | Lower support assembly: lower support column bodies (cast) | Cast austenitic stainless steel | Reactor coolant and neutron flux | Cracking due to ~~irradiation-assisted SCC~~IASCC or fatigue | AMP XI.M16A, "PWR Vessel Internals," and AMP XI.M2, "Water Chemistry" (for SCC mechanisms only) | Yes |
| M | IV.B2.RP-290 | 3.1-1, 059b | Lower support assembly: lower support column bodies (cast) | Cast austenitic stainless steel | Reactor coolant and neutron flux | Loss of fracture toughness due to thermal aging and neutron irradiation embrittlement; changes in dimension due to void swelling or distortion | AMP XI.M16A, "PWR Vessel Internals" | Yes |
| M | IV.B2.RP-294 | 3.1-1, 053b | Lower support assembly: lower support column bodies (non-cast) | Stainless steel | Reactor coolant and neutron flux | Cracking due to ~~irradiation-assisted SCC~~IASCC or fatigue | AMP XI.M16A, "PWR Vessel Internals," and AMP XI.M2, "Water Chemistry" (for SCC mechanisms only) | Yes |

| IV<br>Table B2 | REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM<br>Reactor Vessel Internals (PWR)—Westinghouse | | | | | | | |
|---|---|---|---|---|---|---|---|---|
| **New, Modified, Deleted, Edited Item** | **Item** | **SRP Item (Table, ID)** | **Structure and/or Component** | **Material** | **Environment** | **Aging Effect/Mechanism** | **Aging Management Program (AMP)/TLAA** | **Further Evaluation** |
| M | IV.B2.RP-295 | 3.1-1, 059b | Lower support assembly: lower support column bodies (non-cast) | Stainless steel | Reactor coolant and neutron flux | Loss of fracture toughness due to neutron irradiation embrittlement; changes in dimension due to void swelling or distortion | AMP XI.M16A, "PWR Vessel Internals" | Yes |
| M | IV.B2.RP-286 | 3.1-1, 053b | Lower support assembly: lower support column bolts | Stainless steel, nickel alloy | Reactor coolant and neutron flux | Cracking due to ~~irradiation-assisted SCC~~IASCC or fatigue | AMP XI.M16A, "PWR Vessel Internals," and AMP XI.M2, "Water Chemistry" (for SCC mechanisms only) | Yes |
| M | IV.B2.RP-287 | 3.1-1, 059b | Lower support assembly: lower support column bolts | Stainless steel, nickel alloy | Reactor coolant and neutron flux | Loss of fracture toughness due to neutron irradiation embrittlement; loss of preload due to thermal and irradiation-enhanced stress relaxation or creep; changes in dimension due to void swelling or distortion; loss of material due to wear | AMP XI.M16A, "PWR Vessel Internals" | Yes |

| IV<br>Table B2 | REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM<br>Reactor Vessel Internals (PWR)—Westinghouse | | | | | | | |
|---|---|---|---|---|---|---|---|---|
| **New, Modified, Deleted, Edited Item** | **Item** | **SRP Item (Table, ID)** | **Structure and/or Component** | **Material** | **Environment** | **Aging Effect/Mechanism** | **Aging Management Program (AMP)/TLAA** | **Further Evaluation** |
| N | IV.B2.R-423 | 3.1-1, 118 | Reactor vessel internal components, or specified reactor vessel internal component with a site-specific or component-specific aging management basis | Stainless steel, nickel alloy | Reactor coolant, neutron flux | Cracking due to SCC, ~~irradiation-assisted SCC~~IASCC, cyclic loading, fatigue | Plant-specific aging management program, or AMP XI.M16A, "PWR Vessel Internals," and AMP XI.M2, "Water Chemistry" (SCC and IASCC only), for cases where a specified component is subject to a site-specific or component-specific aging management basis | Yes |
| N | IV.B2.R-424 | 3.1-1, 119 | Reactor vessel internal components, or specified reactor vessel internal component with a site-specific or component-specific aging management basis | Stainless steel, nickel alloy | Reactor coolant, neutron flux | Loss of fracture toughness due to neutron irradiation embrittlement or thermal aging embrittlement; changes in dimensions due to void swelling or distortion; loss of preload due to thermal and irradiation-enhanced stress relaxation or creep; loss of material due to wear | Plant-specific aging management program, or AMP XI.M16A, "PWR Vessel Internals," for cases where a specified component is subject to a site-specific or component-specific aging management basis | Yes |

| IV<br>Table B2 | REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM<br>Reactor Vessel Internals (PWR)—Westinghouse | | | | | | | |
|---|---|---|---|---|---|---|---|---|
| **New, Modified, Deleted, Edited Item** | **Item** | **SRP Item (Table, ID)** | **Structure and/or Component** | **Material** | **Environment** | **Aging Effect/Mechanism** | **Aging Management Program (AMP)/TLAA** | **Further Evaluation** |
| ~~M~~D | IV.B2.RP-382 | ~~3.1-1, 032~~ | ~~Reactor vessel internals: ASME Section XI, Examination Category B-N-3 core support structure components (not already identified as "Existing Programs" components in MRP-227-A)~~ | ~~Stainless steel, nickel alloy, cast austenitic stainless steel~~ | ~~Reactor coolant and neutron flux~~ | ~~Cracking due to fatigue, SCC, or irradiation-assisted SCC; loss of material due to wear~~ | ~~AMP XI.M1, "ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD"~~ | ~~No~~ |
| M | IV.B2.RP-302a | 3.1-1, 059a | Thermal shield assembly: thermal shield flexures | Stainless steel | Reactor coolant and neutron flux | Loss of material due to wear; loss of fracture toughness due to neutron irradiation embrittlement; loss of preload due to irradiation-enhanced stress relaxation or creep | AMP XI.M16A, "PWR Vessel Internals" | Yes |
| M | IV.B2.RP-302 | 3.1-1, 053a | Thermal shield assembly: thermal shield flexures | Stainless steel | Reactor coolant and neutron flux | Cracking due to SCC, IASCC, or fatigue | AMP XI.M16A, "PWR Vessel Internals," and AMP XI.M2, "Water Chemistry" (for SCC mechanisms only) | Yes |

| IV Table B2 | REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM Reactor Vessel Internals (PWR)—Westinghouse | | | | | | | |
|---|---|---|---|---|---|---|---|---|
| **New, Modified, Deleted, Edited Item** | **Item** | **SRP Item (Table, ID)** | **Structure and/or Component** | **Material** | **Environment** | **Aging Effect/Mechanism** | **Aging Management Program (AMP)/TLAA** | **Further Evaluation** |
| M | IV.B2.RP-291b | 3.1-1, 053b | Upper internals assembly; upper core plate | Stainless steel | Reactor coolant and neutron flux | Cracking due to IASCC or fatigue | AMP XI.M16A, "PWR Vessel Internals," and AMP XI.M2, "Water Chemistry" (for SCC mechanisms only) | Yes |
| M | IV.B2.RP-290b | 3.1-1, 059b | Upper internals assembly; upper core plate | Stainless steel | Reactor coolant and neutron flux | Loss of material due to wear; loss of fracture toughness due to neutron irradiation embrittlement | AMP XI.M16A, "PWR Vessel Internals" | Yes |

# APPENDIX B.2

## PROPOSED REVISIONS TO GALL-SLR REPORT TABLE IV.B3, “REACTOR VESSEL INTERNALS (PWR)—COMBUSTION ENGINEERING

NUREG-2191, Volumes 1 and 2, “Generic Aging Lessons Learned for Subsequent License Renewal (GALL-SLR) Report,” Table IV.B3, “Reactor Vessel Internals (PWR)—Combustion Engineering,” addresses the Combustion Engineering (CE) pressurized-water reactor (PWR) vessel internals, which consist of components in the upper internals assembly, the control element assembly (CEA), the core support barrel assembly, the core shroud assembly, and the lower support structure assembly, and incore instrumentation components.

Proposed revisions to Table IV.B3 of the GALL-SLR Report are provided in redline format. These AMR items superseded the respective items in GALL-SLR Report, Revision 0, Table IV.B3.

GALL-SLR Report Table IV.B3 Proposed Revisions

| IV<br>Table B3 | REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM<br>Reactor Vessel Internals (PWR)—Combustion Engineering | | | | | | | |
|---|---|---|---|---|---|---|---|---|
| **New, Modified, Deleted, Edited Item** | **Item** | **SRP Item (Table, ID)** | **Structure and/or Component** | **Material** | **Environment** | **Aging Effect/Mechanism** | **Aging Management Program (AMP)/TLAA** | **Further Evaluation** |
| M | IV.B3.RP-313 | 3.1-1, 052b | Control element assembly (CEA~~):~~ ~~shroud assemblies:~~) - Shroud Assemblies: remaining instrument guide tubes (i.e., guide tubes in non-peripheral ~~CEA~~control element shroud assemblies) | Stainless steel | Reactor coolant and neutron flux | Cracking due to SCC or fatigue | AMP XI.M16A, "PWR Vessel Internals," and AMP XI.M2, "Water Chemistry" (for SCC mechanisms only) | Yes |
| M | IV.B3.RP-312 | 3.1-1, 052a | Control element assembly (CEA~~):~~ ~~shroud assemblies~~) – Shroud Assemblies: instrument guide tubes in peripheral CEA shroud assemblies | Stainless steel | Reactor coolant and neutron flux | Cracking due to SCC or fatigue | AMP XI.M16A, "PWR Vessel Internals," and AMP XI.M2, "Water Chemistry" (for SCC mechanisms only) | Yes |
| M | IV.B3.RP-320 | 3.1-1, 052c | Core shroud and upper internals assemblies ~~(all plants):~~: guide lugs; ~~insert~~guide lug inserts and bolts | Stainless steel | Reactor coolant and neutron flux | Cracking due to fatigue | AMP XI.M16A, "PWR Vessel Internals" | Yes |
| M | IV.B3.RP-319 | 3.1-1, 056c | Core shroud and upper internals assemblies ~~(all plants):~~: guide lugs; ~~insert~~guide lug inserts and bolts | Stainless steel | Reactor coolant and neutron flux | Loss of material due to wear; Loss of preload due to thermal and irradiation-enhanced stress relaxation or creep | AMP XI.M16A, "PWR Vessel Internals" | Yes |

**IV REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM**
**Table B3 Reactor Vessel Internals (PWR)—Combustion Engineering**

| New, Modified, Deleted, Edited Item | Item | SRP Item (Table, ID) | Structure and/or Component | Material | Environment | Aging Effect/Mechanism | Aging Management Program (AMP)/TLAA | Further Evaluation |
|---|---|---|---|---|---|---|---|---|
| M | IV.B3.RP-358 | 3.1-1, 052a | Core shroud assemblies (for bolted core shroud assemblies): assembly components, including core side surfaces, shroud plates and ~~former plates~~plate joints, and bolts and bolt locking devices | Stainless steel | Reactor coolant and neutron flux | Cracking due to ~~irradiation-assisted SCC~~IASCC | AMP XI.M16A, "PWR Vessel Internals," and AMP XI.M2, "Water Chemistry" (for SCC mechanisms only) | Yes |
| M | IV.B3.RP-318 | 3.1-1, 056a | Core shroud assemblies (for bolted core shroud assemblies): assembly components, including core side surfaces, shroud plates and ~~former plates~~plate joints, and bolts and bolt locking devices | Stainless steel | Reactor coolant and neutron flux | Loss of fracture toughness due to neutron irradiation embrittlement; changes in dimensions due to void swelling or distortion | AMP XI.M16A, "PWR Vessel Internals" | Yes |
| M | IV.B3.RP-316 | 3.1-1, 052b | Core shroud assemblies (for bolted core shroud assemblies): barrel-shroud bolts | Stainless steel | Reactor coolant and neutron flux | Cracking due to ~~irradiation-assisted SCC~~IASCC or fatigue | AMP XI.M16A, "PWR Vessel Internals," and AMP XI.M2, "Water Chemistry" (for SCC mechanisms only) | Yes |

| IV<br>Table B3 | REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM<br>Reactor Vessel Internals (PWR)—Combustion Engineering | | | | | | | |
|---|---|---|---|---|---|---|---|---|
| **New, Modified, Deleted, Edited Item** | **Item** | **SRP Item (Table, ID)** | **Structure and/or Component** | **Material** | **Environment** | **Aging Effect/Mechanism** | **Aging Management Program (AMP)/TLAA** | **Further Evaluation** |
| M | IV.B3.RP-314 | 3.1-1, 052a | Core shroud assemblies (for bolted core shroud assemblies): core shroud bolts | Stainless steel | Reactor coolant and neutron flux | Cracking due to ~~irradiation-assisted SCC~~IASCC or fatigue | AMP XI.M16A, "PWR Vessel Internals," and AMP XI.M2, "Water Chemistry" (for SCC mechanisms only) | Yes |
| M | IV.B3.RP-315 | 3.1-1, 056a | Core shroud assemblies (for bolted core shroud assemblies): core shroud bolts | Stainless steel | Reactor coolant and neutron flux | Loss of preload due to thermal and irradiation-enhanced stress relaxation or creep; loss of fracture toughness due to neutron irradiation embrittlement; changes in dimension due to void swelling or distortion | AMP XI.M16A, "PWR Vessel Internals" | Yes |
| M | IV.B3.RP-326 | 3.1-1, 056a | Core shroud assembly (for welded shroud designs assembled in two vertical sections): assembly ~~components,~~ (including ~~monitoring of the gap opening at the core shroud re-entrant corners~~the horizontal seam between the upper and lower shroud segments) | Stainless steel | Reactor coolant and neutron flux | Changes in dimensions due to void swelling or distortion; loss of fracture toughness due to neutron irradiation embrittlement | AMP XI.M16A, "PWR Vessel Internals" | Yes |

**IV REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM**
**Table B3 Reactor Vessel Internals (PWR)—Combustion Engineering**

| New, Modified, Deleted, Edited Item | Item | SRP Item (Table, ID) | Structure and/or Component | Material | Environment | Aging Effect/Mechanism | Aging Management Program (AMP)/TLAA | Further Evaluation |
|---|---|---|---|---|---|---|---|---|
| M | IV.B3.RP-326a | 3.1-1, 052a | Core shroud assembly (for welded shroud designs assembled in two vertical sections): assembly ~~components,~~ (including ~~monitoring of the gap opening at the core shroud re-entrant corners~~the horizontal seam between the upper and lower shroud segments) | Stainless steel | Reactor coolant and neutron flux | Cracking due to ~~SCC~~IASCC or fatigue | AMP XI.M16A, "PWR Vessel Internals," and AMP XI.M2, "Water Chemistry" (for SCC mechanisms only) | Yes |
| M | IV.B3.RP-322 | 3.1-1, 052a | Core shroud assembly (for welded core shroud designs assembled in two vertical sections): core shroud plate-to-former plate welds | Stainless steel | Reactor coolant and neutron flux | Cracking due to ~~irradiation-assisted SCC~~IASCC | AMP XI.M16A, "PWR Vessel Internals," and AMP XI.M2, "Water Chemistry" (for SCC mechanisms only) | Yes |
| M | IV.B3.RP-359 | 3.1-1, 056a | Core shroud assembly (for welded core shroud designs assembled in two vertical sections): core shroud plate-to-former plate welds | Stainless steel | Reactor coolant and neutron flux | Loss of fracture toughness due to neutron irradiation embrittlement; changes in dimensions due to void swelling or distortion | AMP XI.M16A, "PWR Vessel Internals" | Yes |

| IV<br>Table B3 | REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM<br>Reactor Vessel Internals (PWR)—Combustion Engineering | | | | | | | |
|---|---|---|---|---|---|---|---|---|
| **New, Modified, Deleted, Edited Item** | **Item** | **SRP Item (Table, ID)** | **Structure and/or Component** | **Material** | **Environment** | **Aging Effect/Mechanism** | **Aging Management Program (AMP)/TLAA** | **Further Evaluation** |
| M | IV.B3.RP-323 | 3.1-1, 052b | Core shroud assembly (for welded core shroud designs assembled in two vertical sections): remaining axial welds | Stainless steel | Reactor coolant and neutron flux | Cracking due to ~~irradiation-assisted SCC~~IASCC | AMP XI.M16A, "PWR Vessel Internals," and AMP XI.M2, "Water Chemistry" (for SCC mechanisms only) | Yes |
| M | IV.B3.RP-359a | 3.1-1, 056b | Core shroud assembly (for welded core shroud designs assembled in two vertical sections): remaining axial welds | Stainless steel | Reactor coolant and neutron flux | Loss of fracture toughness due to neutron irradiation embrittlement; ~~changes in dimensions due to void swelling or distortion~~ | AMP XI.M16A, "PWR Vessel Internals" | Yes |
| M | IV.B3.RP-325 | 3.1-1, 052b | Core shroud assembly (for core shroud designs assembled with full-height shroud plates): remaining axial welds, ribs, and rings | Stainless steel | Reactor coolant and neutron flux | Cracking due to ~~irradiation-assisted SCC~~IASCC | AMP XI.M16A, "PWR Vessel Internals," and AMP XI.M2, "Water Chemistry" (for SCC mechanisms only) | Yes |
| M | IV.B3.RP-361 | 3.1-1, 056b | Core shroud assembly (for core shroud designs assembled with full-height shroud plates): remaining axial welds, ribs, and rings | Stainless steel | Reactor coolant and neutron flux | Loss of fracture toughness due to neutron irradiation embrittlement | AMP XI.M16A, "PWR Vessel Internals" | Yes |

| IV Table B3 | REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM Reactor Vessel Internals (PWR)—Combustion Engineering | | | | | | | |
|---|---|---|---|---|---|---|---|---|
| **New, Modified, Deleted, Edited Item** | **Item** | **SRP Item (Table, ID)** | **Structure and/or Component** | **Material** | **Environment** | **Aging Effect/Mechanism** | **Aging Management Program (AMP)/TLAA** | **Further Evaluation** |
| M | IV.B3.RP-360 | 3.1-1, 056a | Core shroud assembly (for core shroud designs assembled with full-height shroud plates): shroud ~~plate~~plates (including visible axial weld seams at the core shroud re-entrant corners and at the core midplane) | Stainless steel | Reactor coolant and neutron flux | Loss of fracture toughness due to neutron irradiation embrittlement; changes in dimension due to void swelling or distortion | AMP XI.M16A, "PWR Vessel Internals" | Yes |
| M | IV.B3.RP-324 | 3.1-1, 052a | Core shroud assembly (core shroud designs assembled with full-height shroud plates): shroud plates~~,~~ (including visible axial weld seams at the core shroud re-entrant corners~~,~~ and at the core mid-plane ~~(+3 feet in height) as visible from the core side of the shroud~~) | Stainless steel | Reactor coolant and neutron flux | Cracking due to ~~irradiation-assisted SCC~~IASCC | AMP XI.M16A, "PWR Vessel Internals," and AMP XI.M2, "Water Chemistry" (for SCC mechanisms only) | Yes |
| M | IV.B3.RP-328 | 3.1-1, 052a | Core support barrel assembly: ~~lower core barrel flange~~flexure weld | Stainless steel | Reactor coolant and neutron flux | Cracking due to SCC or fatigue | AMP XI.M16A, "PWR Vessel Internals," and AMP XI.M2, "Water Chemistry" (for SCC mechanisms only) | Yes |

| IV<br>Table B3 | REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM<br>Reactor Vessel Internals (PWR)—Combustion Engineering | | | | | | | |
|---|---|---|---|---|---|---|---|---|
| **New, Modified, Deleted, Edited Item** | **Item** | **SRP Item (Table, ID)** | **Structure and/or Component** | **Material** | **Environment** | **Aging Effect/Mechanism** | **Aging Management Program (AMP)/TLAA** | **Further Evaluation** |
| M | IV.B3.RP-362 | 3.1-1, 056a | Core support barrel assembly: ~~lower cylinder~~middle circumferential (girth) ~~welds~~weld | Stainless steel | Reactor coolant and neutron flux | Loss of fracture toughness due to neutron irradiation embrittlement | AMP XI.M16A, "PWR Vessel Internals" | Yes |
| M | IV.B3.RP-362a | 3.1-1, 052a | Core support barrel assembly: ~~lower cylinder~~middle circumferential (girth) ~~welds~~weld | Stainless steel | Reactor coolant and neutron flux | Cracking due to SCC or ~~irradiation-assisted SCC~~IASCC | AMP XI.M16A, "PWR Vessel Internals," and AMP XI.M2, "Water Chemistry" (for SCC mechanisms only) | Yes |
| M | IV.B3.RP-362c | 3.1-1, 052b | Core support barrel assembly: ~~lower cylinder~~middle vertical (axial) welds and lower vertical (axial) welds | Stainless steel | Reactor coolant and neutron flux | Cracking due to SCC or ~~irradiation-assisted SCC~~IASCC | AMP XI.M16A, "PWR Vessel Internals," and AMP XI.M2, "Water Chemistry" (for SCC mechanisms only) | Yes |
| M | IV.B3.RP-362b | 3.1-1, 056b | Core support barrel assembly: ~~lower cylinder~~middle vertical (axial) welds and lower vertical (axial) welds | Stainless steel | Reactor coolant and neutron flux | Loss of fracture toughness due to neutron irradiation embrittlement | AMP XI.M16A, "PWR Vessel Internals" | Yes |
| M | IV.B3.RP-333 | 3.1-1, 052b | Core support barrel assembly: lower girth weld (lower flange weld) | Stainless steel | Reactor coolant and neutron flux | Cracking due to SCC, IASCC, or fatigue | AMP XI.M16A, "PWR Vessel Internals," and AMP XI.M2, "Water Chemistry" (for SCC mechanisms only) | Yes |

| IV<br>Table B3 | REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM<br>Reactor Vessel Internals (PWR)—Combustion Engineering | | | | | | | |
|---|---|---|---|---|---|---|---|---|
| **New, Modified, Deleted, Edited Item** | **Item** | **SRP Item (Table, ID)** | **Structure and/or Component** | **Material** | **Environment** | **Aging Effect/Mechanism** | **Aging Management Program (AMP)/TLAA** | **Further Evaluation** |
| N | IV.B3.RP-333a | 3.1-1, 056b | Core support barrel assembly: lower girth weld (lower flange weld) | Stainless steel | Reactor coolant and neutron flux | Loss of fracture toughness due to neutron irradiation embrittlement | AMP XI.M16A, "PWR Vessel Internals" | Yes |
| ~~M~~D | IV.B3.RP-400 | ~~3.1-1, 028~~ | ~~Core support barrel assembly: thermal shield positioning pins~~ | ~~Stainless steel~~ | ~~Reactor coolant and neutron flux~~ | ~~Cracking due to SCC, irradiation-assisted SCC or fatigue; loss of material due to wear~~ | ~~AMP XI.M16A, "PWR Vessel Internals," and AMP XI.M2, "Water Chemistry" (for SCC mechanisms only)~~ | ~~Yes~~ |
| M | IV.B3.RP-332 | 3.1-1, 056c | Core support barrel assembly: upper ~~core barrel~~ flange | Stainless steel | Reactor coolant and neutron flux | Loss of material due to wear | AMP XI.M16A, "PWR Vessel Internals" | Yes |
| M | IV.B3.RP-327 | 3.1-1, 052a | Core support barrel assembly: upper ~~core support barrel~~ flange weld | Stainless steel | Reactor coolant and neutron flux | Cracking due to SCC | AMP XI.M16A, "PWR Vessel Internals," and AMP XI.M2, "Water Chemistry" (for SCC mechanisms only) | Yes |
| N | IV.B3.R-455 | 3.1-1, 056b | Core support barrel assembly: upper ~~cylinder (base metal~~circumferential (girth) weld and upper vertical (axial) welds~~)~~ | Stainless steel | Reactor coolant and neutron flux | Loss of fracture toughness due to neutron irradiation embrittlement | AMP XI.M16A, "PWR Vessel Internals," and AMP XI.M2, "Water Chemistry" (for SCC mechanisms only) | Yes |

| IV<br>Table B3 | REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM<br>Reactor Vessel Internals (PWR)—Combustion Engineering | | | | | | | |
|---|---|---|---|---|---|---|---|---|
| **New, Modified, Deleted, Edited Item** | **Item** | **SRP Item (Table, ID)** | **Structure and/or Component** | **Material** | **Environment** | **Aging Effect/Mechanism** | **Aging Management Program (AMP)/TLAA** | **Further Evaluation** |
| M | IV.B3.RP-329 | 3.1-1, 052b | Core support barrel assembly: upper ~~cylinder (base metal and welds)~~circumferential (girth) weld and upper ~~core barrel flange (flange base metal)~~vertical (axial) welds | Stainless steel | Reactor coolant and neutron flux | Cracking due to SCC | AMP XI.M16A, "PWR Vessel Internals," and AMP XI.M2, "Water Chemistry" (for SCC mechanisms only) | Yes |
| M | IV.B3.RP-357 | 3.1-1, ~~028~~056c | Incore instruments (ICI): ICI thimble tubes - lower | Zircaloy-4 | Reactor coolant and neutron flux | Loss of material due to wear | AMP XI.M16A, "PWR Vessel Internals" | Yes |
| M | IV.B3.RP-363 | 3.1-1, ~~052a~~052b | Lower support structure (all plants with either full height bolted or half height welded shroud plates): core support ~~column welds~~columns | Stainless steel | Reactor coolant and neutron flux | Cracking due to SCC, ~~irradiation-assisted SCC~~IASCC, or fatigue | AMP XI.M16A, "PWR Vessel Internals," and AMP XI.M2, "Water Chemistry" (for SCC mechanisms only) | Yes |
| M | IV.B3.RP-364 | 3.1-1, ~~056a~~056b | Lower support structure (all plants with either full height bolted or half height welded shroud plates): core support ~~column welds~~columns | Stainless steel (including CASS) | Reactor coolant and neutron flux | Loss of fracture toughness due to neutron irradiation and thermal embrittlement (TE for CASS materials only) | AMP XI.M16A, "PWR Vessel Internals" | Yes |

| **IV Table B3** | **REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM Reactor Vessel Internals (PWR)—Combustion Engineering** | | | | | | | |
|---|---|---|---|---|---|---|---|---|
| **New, Modified, Deleted, Edited Item** | **Item** | **SRP Item (Table, ID)** | **Structure and/or Component** | **Material** | **Environment** | **Aging Effect/Mechanism** | **Aging Management Program (AMP)/TLAA** | **Further Evaluation** |
| M | IV.B3.RP-334 | 3.1-1, 052c | Lower support structure (for CE plants with core shroud designs assembled in two vertical sections or ~~with~~from full-height shroud plates): fuel alignment pins | Stainless steel | Reactor coolant and neutron flux | Cracking due to SCC, ~~irradiation-assisted SCC~~IASCC, or fatigue | AMP XI.M16A, "PWR Vessel Internals," and AMP XI.M2, "Water Chemistry" (for SCC mechanisms only) | Yes |
| M | IV.B3.RP-336 | 3.1-1, 056c | Lower support structure (for CE plants with core shroud designs assembled in two vertical sections or from full height shroud plates): fuel alignment pins | Stainless steel | Reactor coolant and neutron flux | Loss of material due to wear; loss of fracture toughness due to neutron irradiation embrittlement; loss of preload due to thermal and irradiation-enhanced stress relaxation or creep | AMP XI.M16A, "PWR Vessel Internals" | Yes |
| ~~M~~D | IV.B3.RP-334a | ~~3.1-1, 056c~~ | ~~Lower support structure (designs assembled with full-height shroud plates): fuel alignment pins~~ | ~~Stainless steel~~ | ~~Reactor coolant and neutron flux~~ | ~~Loss of material due to wear; loss of fracture toughness due to neutron irradiation embrittlement; loss of preload due to thermal and irradiation-enhanced stress relaxation or creep~~ | ~~AMP XI.M16A, "PWR Vessel Internals"~~ | ~~Yes~~ |

| IV<br>Table B3 | REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM<br>Reactor Vessel Internals (PWR)—Combustion Engineering | | | | | | |
|---|---|---|---|---|---|---|---|
| **New, Modified, Deleted, Edited Item** | **Item** | **SRP Item (Table, ID)** | **Structure and/or Component** | **Material** | **Environment** | **Aging Effect/Mechanism** | **Aging Management Program (AMP)/TLAA** | **Further Evaluation** |
| M | IV.B3.RP-335 | 3.1-1, 052b | Lower support structure (all CE plants except those with welded core shroud designs assembled ~~with~~from full-height shroud plates): lower core support beams | Stainless steel | Reactor coolant and neutron flux | Cracking due to SCC~~, irradiation-assisted SCC,~~ or fatigue | AMP XI.M16A, "PWR Vessel Internals," and AMP XI.M2, "Water Chemistry" (for SCC mechanisms only) | Yes |
| M | IV.B3.RP-343 | 3.1-1, 052a | Lower support structure (for CE plant designs with a core support plate): core support plate | Stainless steel | Reactor coolant and neutron flux | Cracking due to fatigue | AMP XI.M16A, "PWR Vessel Internals," and AMP XI.M2, "Water Chemistry" (for SCC mechanisms only) | Yes |
| M | IV.B3.RP-365 | 3.1-1, 056a | Lower support structure (for CE plant designs with a core support plate): core support plate | Stainless steel | Reactor coolant and neutron flux | Loss of fracture toughness due to neutron irradiation embrittlement | AMP XI.M16A, "PWR Vessel Internals" | Yes |
| M | IV.B3.RP-342 | 3.1-1, 052a | Lower support structure (~~designs~~ for CE plants with welded core ~~shrouds~~ shroud designs assembled ~~with~~from full height shroud plates): deep beams | Stainless steel | Reactor coolant and neutron flux | Cracking due to SCC, ~~irradiation-assisted SCC~~IASCC, or fatigue | AMP XI.M16A, "PWR Vessel Internals," and AMP XI.M2, "Water Chemistry" (for SCC mechanisms only) | Yes |

| IV<br>Table B3 | REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM<br>Reactor Vessel Internals (PWR)—Combustion Engineering | | | | | | | |
|---|---|---|---|---|---|---|---|---|
| **New, Modified, Deleted, Edited Item** | **Item** | **SRP Item (Table, ID)** | **Structure and/or Component** | **Material** | **Environment** | **Aging Effect/Mechanism** | **Aging Management Program (AMP)/TLAA** | **Further Evaluation** |
| M | IV.B3.RP-366 | 3.1-1, 056a | Lower support structure (for CE plants with welded core shroud designs ~~with~~ assembled from full height shroud plates): deep beams | Stainless steel | Reactor coolant and neutron flux | Loss of fracture toughness due to neutron irradiation embrittlement | AMP XI.M16A, "PWR Vessel Internals" | Yes |
| M | IV.B3.RP-330 | 3.1-1, 052b | Lower support structure~~:~~ (for CE plants with bolted designs): core support column bolts | Stainless steel | Reactor coolant and neutron flux | Cracking due to ~~irradiation-assisted SCC~~IASCC or fatigue | AMP XI.M16A, "PWR Vessel Internals," and AMP XI.M2, "Water Chemistry" (for SCC mechanisms only) | Yes |
| M | IV.B3.RP-331 | 3.1-1, 056b | Lower support structure~~:~~ (for CE plants with bolted designs): core support column bolts | Stainless steel | Reactor coolant and neutron flux | Loss of fracture toughness due to neutron irradiation embrittlement | AMP XI.M16A, "PWR Vessel Internals" | Yes |

| IV<br>Table B3 | REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM<br>Reactor Vessel Internals (PWR)—Combustion Engineering | | | | | | | |
|---|---|---|---|---|---|---|---|---|
| **New, Modified, Deleted, Edited Item** | **Item** | **SRP Item (Table, ID)** | **Structure and/or Component** | **Material** | **Environment** | **Aging Effect/Mechanism** | **Aging Management Program (AMP)/TLAA** | **Further Evaluation** |
| N | IV.B3.R-423 | 3.1-1, 118 | Reactor vessel internal components or reactor vessel internal component-specific basis for a specified RVI component | Stainless steel, nickel alloy | Reactor coolant, neutron flux | Cracking due to SCC, ~~irradiation-assisted SCC~~IASCC, cyclic loading, fatigue | Plant-specific aging management program, or AMP XI.M16A, "PWR Vessel Internals," and AMP XI.M2, "Water Chemistry" (SCC and IASCC only), for cases where a specified component is subject to a site-specific or component-specific aging management basis | Yes |

| IV<br>Table B3 | REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM<br>Reactor Vessel Internals (PWR)—Combustion Engineering | | | | | | | |
|---|---|---|---|---|---|---|---|---|
| **New, Modified, Deleted, Edited Item** | **Item** | **SRP Item (Table, ID)** | **Structure and/or Component** | **Material** | **Environment** | **Aging Effect/Mechanism** | **Aging Management Program (AMP)/TLAA** | **Further Evaluation** |
| N | IV.B3.R-424 | 3.1-1, 119 | Reactor vessel internal components or reactor vessel internal component-specific basis for a specified RVI component | Stainless steel, nickel alloy | Reactor coolant, neutron flux | Loss of fracture toughness due to neutron irradiation embrittlement or thermal aging embrittlement; changes in dimensions due to void swelling or distortion; loss of preload due to thermal and irradiation-enhanced stress relaxation or creep; loss of material due to wear | Plant-specific aging management program, or AMP XI.M16A, "PWR Vessel Internals," for cases where a specified component is subject to a site-specific or component-specific aging management basis | Yes |
| ~~M~~D | IV.B3.RP-382 | ~~3.1-1, 032~~ | ~~Reactor vessel internals: ASME Section XI, Examination Category B-N-3 core support structure components (not already identified as "Existing Programs" components in MRP-227-A)~~ | ~~Stainless steel, nickel alloy, cast austenitic stainless steel~~ | ~~Reactor coolant and neutron flux~~ | ~~Cracking due to fatigue, SCC, or irradiation-assisted SCC; loss of material due to wear~~ | ~~AMP XI.M1, "ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD"~~ | ~~No~~ |
| M | IV.B3.RP-338 | 3.1-1, 052a | Upper internals assembly (~~designs~~ for CE plants with core ~~shrouds~~ shroud designs assembled ~~with~~from full height shroud plates): fuel alignment plate | Stainless steel | Reactor coolant and neutron flux | Cracking due to fatigue | AMP XI.M16A, "PWR Vessel Internals" | Yes |

| IV<br>Table B3 | REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM<br>Reactor Vessel Internals (PWR)—Combustion Engineering | | | | | | | |
|---|---|---|---|---|---|---|---|---|
| **New, Modified, Deleted, Edited Item** | **Item** | **SRP Item (Table, ID)** | **Structure and/or Component** | **Material** | **Environment** | **Aging Effect/Mechanism** | **Aging Management Program (AMP)/TLAA** | **Further Evaluation** |
| N | IV.B3.RP-338a | 3.1-1, 056a | Upper internals assembly (for CE plants with core shroud designs assembled from full height shroud plates): fuel alignment plate | Stainless steel | Reactor coolant and neutron flux | Loss of fracture toughness due to neutron irradiation embrittlement | AMP XI.M16A, "PWR Vessel Internals" | Yes |
| N | IV.B3.RP-320a | 3.1-1, 052c | Alignment and Interfacing Components: core stabilizing lugs, shims and bolts | Stainless, steel, nickel alloy | Reactor coolant and neutron flux | Cracking due to SCC | AMP XI.M16A, "PWR Vessel Internals" | Yes |

# APPENDIX B.3

## PROPOSED REVISIONS TO GALL-SLR REPORT TABLE IV.B4, “REACTOR VESSEL INTERNALS (PWR)—BABCOCK & WILCOX”

NUREG-2191, Volumes 1 and 2, “Generic Aging Lessons Learned for Subsequent License Renewal (GALL-SLR) Report,” Table IV.B4, “Reactor Vessel Internals (PWR)—Babcock & Wilcox,” addresses the Babcock & Wilcox (B&W) pressurized-water reactor (PWR) vessel internals, which consist of components in the plenum cover assembly, the upper grid assembly, the control rod guide tube (CRGT) assembly, the core support shield assembly, the core barrel assembly, the lower grid assembly, the incore monitoring instrument (IMI) guide tube assembly, and the flow distributor assembly.

Proposed revisions to Table IV.B4 of the GALL-SLR Report are provided in redline format. These AMR items supersede the respective items in GALL-SLR Report, Revision 0, Table IV.B4.

GALL-SLR Report Table IV.B4 Proposed Revisions

| IV Table B4 | REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM Reactor Vessel Internals (PWR)—Babcock & Wilcox | | | | | | | |
|---|---|---|---|---|---|---|---|---|
| **New, Modified, Deleted, Edited Item** | **Item** | **SRP Item (Table, ID)** | **Structure and/or Component** | **Material** | **Environment** | **Aging Effect/Mechanism** | **Aging Management Program (AMP)/TLAA** | **Further Evaluation** |
| M | IV.B4.RP-245 | 3.1-1, 051b | Core barrel assembly (applicable to Davis Besse only): surveillance specimen holder tube (SSHT) studs~~/nuts~~ or bolts | Stainless steel, nickel alloy | Reactor coolant and neutron flux | Cracking due to SCC | AMP XI.M16A, "PWR Vessel Internals," and AMP XI.M2, "Water Chemistry" | Yes |
| N | IV.B4.RP-245c | 3.1-1, 058b | Core barrel assembly (applicable to Davis Besse only): surveillance specimen holder tube (SSHT) studs or bolts | Stainless steel, nickel alloy | Reactor coolant and neutron flux | Loss of material due to wear; loss of preload due to thermal or irradiation-enhanced stress relaxation or creep | AMP XI.M16A, "PWR Vessel Internals" | Yes |
| M | IV.B4.RP-247 | 3.1-1, 051a | Core barrel assembly: ~~accessible~~ lower core barrel (LCB) bolts~~ and locking devices~~ | Stainless steel, nickel alloy | Reactor coolant and neutron flux | Cracking due to SCC or fatigue | AMP XI.M16A, "PWR Vessel Internals," and AMP XI.M2, "Water Chemistry" | Yes |
| N | IV.B4.RP-247c | 3.1-1, 058a | Core barrel assembly: lower core barrel (LCB) bolts | Stainless steel, nickel alloy | Reactor coolant and neutron flux | Loss of material due to wear; loss of preload due to thermal and irradiation-enhanced stress relaxation or creep | AMP XI.M16A, "PWR Vessel Internals" | Yes |
| M | IV.B4.RP-249a | 3.1-1, 051a | Core barrel assembly: baffle plates | Stainless steel | Reactor coolant and neutron flux | Cracking due to ~~SCC, irradiation-assisted SCC, cyclic loading,~~IASCC or fatigue | AMP XI.M16A, "PWR Vessel Internals," and AMP XI.M2, "Water Chemistry" | Yes |

| **IV<br>Table B4** | **REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM<br>Reactor Vessel Internals (PWR)—Babcock & Wilcox** | | | | | | | |
|---|---|---|---|---|---|---|---|---|
| **New, Modified, Deleted, Edited Item** | **Item** | **SRP Item (Table, ID)** | **Structure and/or Component** | **Material** | **Environment** | **Aging Effect/Mechanism** | **Aging Management Program (AMP)/TLAA** | **Further Evaluation** |
| M | IV.B4.RP-241 | 3.1-1, 051a | Core barrel assembly: baffle/former assembly: baffle-to-former bolts ~~and screws~~ | Stainless steel | Reactor coolant and neutron flux | Cracking due to ~~SCC, irradiation-assisted SCC~~IASCC, fatigue, or overload | AMP XI.M16A, "PWR Vessel Internals," and AMP XI.M2, "Water Chemistry" (for SCC mechanisms only) | Yes |
| M | IV.B4.RP-240 | 3.1-1, 058a | Core barrel assembly: baffle-to-former bolts ~~and screws~~ | Stainless steel | Reactor coolant and neutron flux | Loss of fracture toughness due to neutron irradiation embrittlement; loss of preload due to thermal and irradiation-enhanced stress relaxation; loss of material due to wear | AMP XI.M16A, "PWR Vessel Internals" | Yes |
| M | IV.B4.RP-250a | 3.1-1, 051b | Core barrel assembly: core barrel cylinder (including vertical and circumferential seam welds); former plates | Stainless steel | Reactor coolant and neutron flux | Cracking due to ~~irradiation-assisted SCC~~IASCC or fatigue | AMP XI.M16A, "PWR Vessel Internals," and AMP XI.M2, "Water Chemistry" (~~irradiation-assisted SCC~~IASCC only) | Yes |

**IV REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM**
**Table B4 Reactor Vessel Internals (PWR)—Babcock & Wilcox**

| New, Modified, Deleted, Edited Item | Item | SRP Item (Table, ID) | Structure and/or Component | Material | Environment | Aging Effect/Mechanism | Aging Management Program (AMP)/TLAA | Further Evaluation |
|---|---|---|---|---|---|---|---|---|
| M | IV.B4.RP-244 | 3.1-1, 051b | Core barrel assembly: ~~external and internal~~ baffle-to-baffle bolts and core barrel-to-former bolts | Stainless steel | Reactor coolant and neutron flux | Cracking due to ~~irradiation-assisted SCC~~IASCC, fatigue, or overload | AMP XI.M16A, "PWR Vessel Internals," and AMP XI.M2, "Water Chemistry" (~~irradiation-assisted SCC~~IASCC only) | Yes |
| M | IV.B4.RP-243 | 3.1-1, 058b | Core barrel assembly: ~~external and internal~~ baffle-to-baffle bolts and core barrel-to-former bolts | Stainless steel | Reactor coolant and neutron flux | Loss of fracture toughness due to neutron irradiation embrittlement; loss of preload due to thermal and irradiation-enhanced stress relaxation or creep; loss of material due to wear | AMP XI.M16A, "PWR Vessel Internals" | Yes |
| M | IV.B4.RP-240a | 3.1-1, 058a | Core barrel assembly: locking devices (including locking 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~~; loss of material due to wear~~ | AMP XI.M16A, "PWR Vessel Internals" | Yes |
| M | IV.B4.RP-241a | 3.1-1, 051a | Core barrel assembly: locking devices (including locking welds) of baffle-to-former bolts and internal baffle-to-baffle bolts | Stainless steel | Reactor coolant and neutron flux | Cracking due to ~~SCC, irradiation-assisted SCC~~IASCC~~, fatigue, or overload~~ | AMP XI.M16A, "PWR Vessel Internals," and AMP XI.M2, "Water Chemistry" (for SCC mechanisms only) | Yes |

| **IV**<br>**Table B4** | **REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM**<br>**Reactor Vessel Internals (PWR)—Babcock & Wilcox** | | | | | | | |
|---|---|---|---|---|---|---|---|---|
| **New, Modified, Deleted, Edited Item** | **Item** | **SRP Item (Table, ID)** | **Structure and/or Component** | **Material** | **Environment** | **Aging Effect/Mechanism** | **Aging Management Program (AMP)/TLAA** | **Further Evaluation** |
| M | IV.B4.RP-244a | 3.1-1, 051b | Core barrel assembly: locking devices (including locking welds) of external baffle-to-baffle bolts and core barrel-to-former bolts | Stainless steel | Reactor coolant and neutron flux | Cracking due to ~~irradiation-assisted SCC~~IASCC or fatigue | AMP XI.M16A, "PWR Vessel Internals," and AMP XI.M2, "Water Chemistry" (~~irradiation-assisted SCC~~IASCC only) | Yes |
| M | IV.B4.RP-243a | 3.1-1, 058b | Core barrel assembly: locking devices (including locking welds) of external baffle-to-baffle bolts and core barrel-to-former bolts | Stainless steel | Reactor coolant and neutron flux | Loss of fracture toughness due to neutron irradiation embrittlement~~; loss of material due to wear~~ | AMP XI.M16A, "PWR Vessel Internals" | Yes |
| M | IV.B4.RP-248 | 3.1-1, 051a | Core support shield (CSS) assembly: ~~accessible~~ upper core barrel (UCB) bolts ~~and locking devices~~ | Stainless steel, nickel alloy | Reactor coolant and neutron flux | Cracking due to SCC or fatigue | AMP XI.M16A, "PWR Vessel Internals," and AMP XI.M2, "Water Chemistry" (SCC only) | Yes |
| M | IV.B4.RP-252 | 3.1-1, 058a | ~~Core support shield (CSS)~~ Vent valve assembly: ~~CSS~~ vent valve top and bottom retaining rings ~~(valve body components)~~ | Stainless steel, including CASS and PH steels | Reactor coolant and neutron flux | Loss of fracture toughness due to thermal aging embrittlement | AMP XI.M16A, "PWR Vessel Internals" | Yes |

| IV<br>Table B4 | REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM<br>Reactor Vessel Internals (PWR)—Babcock & Wilcox | | | | | | | |
|---|---|---|---|---|---|---|---|---|
| **New, Modified, Deleted, Edited Item** | **Item** | **SRP Item (Table, ID)** | **Structure and/or Component** | **Material** | **Environment** | **Aging Effect/Mechanism** | **Aging Management Program (AMP)/TLAA** | **Further Evaluation** |
| M | IV.B4.RP-252a | 3.1-1, 051a | ~~Core support shield (CSS)~~ Vent valve assembly: ~~CSS~~ vent valve top and bottom retaining rings~~; vent valve locking devices (valve body components)~~ | Stainless steel | Reactor coolant and neutron flux | Cracking due to SCC or fatigue | AMP XI.M16A, "PWR Vessel Internals," and AMP XI.M2, "Water Chemistry" (for SCC mechanisms only) | Yes |
| N | IV.B4.RP-252b | 3.1-1, 051b | Vent valve assembly: vent valve bodies | CASS | Reactor coolant and neutron flux | Cracking due to SCC or fatigue | AMP XI.M16A, "PWR Vessel Internals," and AMP XI.M2, "Water Chemistry" (SCC only) | Yes |
| N | IV.B4.RP-252c | 3.1-1, 058b | Vent valve assembly: vent valve bodies | CASS | Reactor coolant and neutron flux | Loss of fracture toughness due to thermal aging embrittlement | AMP XI.M16A, "PWR Vessel Internals" | Yes |

| IV Table B4 | REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM Reactor Vessel Internals (PWR)—Babcock & Wilcox | | | | | | | |
|---|---|---|---|---|---|---|---|---|
| **New, Modified, Deleted, Edited Item** | **Item** | **SRP Item (Table, ID)** | **Structure and/or Component** | **Material** | **Environment** | **Aging Effect/Mechanism** | **Aging Management Program (AMP)/TLAA** | **Further Evaluation** |
| N | IV.B4.RP-252d | 3.1-1, 058a | Vent valve assembly: original locking devices (associated with the pressure plate, spring retainer, spring, U-cover, key ring, and pin in the assembly) | Stainless steel | Reactor coolant and neutron flux | Loss of material due to wear (for locking devices associated with the pressure plate, spring and spring retainer, and U cover in the assembly);<br><br>loss of fracture toughness due to thermal aging embrittlement (for locking devices associated with the key ring and pin in the assembly) | AMP XI.M16A, "PWR Vessel Internals" | Yes |
| N | IV.B4.RP-252e | 3.1-1, 051a | Vent valve assembly: original locking devices (associated with the key ring, pin in the assembly); modified locking devices (associated with lock cup, jackscrew locking cup and bolted block in the assembly - Oconee 1, 2, and 3 and ANO-1 only) | Stainless steel | Reactor coolant and neutron flux | Cracking due to SCC or fatigue (fatigue only for listed original locking devices) | AMP XI.M16A, "PWR Vessel Internals," and AMP XI.M2, "Water Chemistry" (for SCC mechanisms only) | Yes |

**IV REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM**
**Table B4 Reactor Vessel Internals (PWR)—Babcock & Wilcox**

| New, Modified, Deleted, Edited Item | Item | SRP Item (Table, ID) | Structure and/or Component | Material | Environment | Aging Effect/Mechanism | Aging Management Program (AMP)/TLAA | Further Evaluation |
|---|---|---|---|---|---|---|---|---|
| ~~M~~D | IV.B4.RP-400 | ~~3.1-1, 051a~~ | ~~Core support shield assembly: upper (top) flange weld~~ | ~~Stainless steel~~ | ~~Reactor coolant and neutron flux~~ | ~~Cracking due to SCC~~ | ~~AMP XI.M16A, "PWR Vessel Internals," and AMP XI.M2, "Water Chemistry"~~ | ~~Yes~~ |
| ~~M~~D | IV.B4.RP-401 | ~~3.1-1, 058a~~ | ~~Core support shield assembly: upper (top) flange weld~~ | ~~Stainless steel~~ | ~~Reactor coolant and neutron flux~~ | ~~Loss of fracture toughness due to neutron irradiation embrittlement~~ | ~~AMP XI.M16A, "PWR Vessel Internals"~~ | ~~Yes~~ |
| M | IV.B4.RP-256a | 3.1-1, 051a | Flow distributor assembly: flow distributor (FD) bolt locking devices | Stainless steel, nickel alloy | Reactor coolant and neutron flux | Cracking due to fatigue | AMP XI.M16A, "PWR Vessel Internals" | Yes |
| M | IV.B4.RP-256b | 3.1-1, 058a | Flow distributor assembly: flow distributor (FD) bolt locking devices | Stainless steel, nickel alloy | Reactor coolant and neutron flux | Loss of material due to wear; changes in dimensions due to distortion or void swelling or distortion | AMP XI.M16A, "PWR Vessel Internals" | Yes |
| M | IV.B4.RP-256 | 3.1-1, 051a | Flow distributor assembly: flow distributor (FD) bolts | Stainless steel, nickel alloy | Reactor coolant and neutron flux | Cracking due to SCC or fatigue | AMP XI.M16A, "PWR Vessel Internals," and AMP XI.M2, "Water Chemistry" | Yes |
| M | IV.B4.RP-258a | 3.1-1, 051a | Incore Monitoring Instrument (IMI) guide tube assembly: IMI guide tube spiders (castings) | ~~Stainless~~Cast austenitic stainless steel | Reactor coolant and neutron flux | Cracking due to SCC, ~~irradiation-assisted SCC~~IASCC, or fatigue | AMP XI.M16A, "PWR Vessel Internals," and AMP XI.M2, "Water Chemistry" (SCC and ~~irradiation-assisted SCC~~IASCC only) | Yes |

| IV<br>Table B4 | REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM<br>Reactor Vessel Internals (PWR)—Babcock & Wilcox | | | | | | | |
|---|---|---|---|---|---|---|---|---|
| **New, Modified, Deleted, Edited Item** | **Item** | **SRP Item (Table, ID)** | **Structure and/or Component** | **Material** | **Environment** | **Aging Effect/Mechanism** | **Aging Management Program (AMP)/TLAA** | **Further Evaluation** |
| M | IV.B4.RP-259a | 3.1-1, 051a | Incore Monitoring Instrument (IMI) guide tube assembly: IMI guide tube spider-to-lower grid rib ~~sections~~section welds | Stainless steel | Reactor coolant and neutron flux | Cracking due to SCC, ~~irradiation-assisted SCC~~IASCC, or fatigue | AMP XI.M16A, "PWR Vessel Internals," and AMP XI.M2, "Water Chemistry" (SCC and ~~irradiation-assisted SCC~~IASCC only) | Yes |
| M | IV.B4.RP-259 | 3.1-1, 058a | Incore Monitoring Instrument (IMI) guide tube assembly: IMI guide tube spider-to-lower grid rib ~~sections~~section welds | Stainless steel~~, nickel alloy~~ | Reactor coolant and neutron flux | Loss of fracture toughness due to ~~thermal aging,~~ neutron irradiation embrittlement | AMP XI.M16A, "PWR Vessel Internals" | Yes |
| M | IV.B4.RP-262 | 3.1-1, 051b | Lower grid assembly: ~~accessible alloy X-750~~ dowel-to-lower grid fuel assembly support pad ~~locking~~ welds (all plants except Davis Besse) | Nickel alloy | Reactor coolant and neutron flux | Cracking due to SCC | AMP XI.M16A, "PWR Vessel Internals," and AMP XI.M2, "Water Chemistry" (for SCC mechanisms only) | Yes |
| M | IV.B4.RP-261 | 3.1-1, 051a | Lower grid assembly: ~~alloy X-750~~ dowel-to-guide block welds (all plants except Davis Besse) | Nickel alloy | Reactor coolant and neutron flux | Cracking due to SCC | AMP XI.M16A, "PWR Vessel Internals," and AMP XI.M2, "Water Chemistry" | Yes |
| ~~M~~D | IV.B4.RP-254b | ~~3.1-1, 058b~~ | ~~Lower grid assembly: alloy X-750 lower grid shock pad bolt locking devices (Three Mile Island Unit 1 only)~~ | ~~Nickel Alloy~~ | ~~Reactor coolant and neutron flux~~ | ~~Loss of material due to wear; changes in dimensions due to void swelling or distortion~~ | ~~AMP XI.M16A, "PWR Vessel Internals"~~ | ~~Yes~~ |

| IV<br>Table B4 | REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM<br>Reactor Vessel Internals (PWR)—Babcock & Wilcox | | | | | | | |
|---|---|---|---|---|---|---|---|---|
| **New, Modified, Deleted, Edited Item** | **Item** | **SRP Item (Table, ID)** | **Structure and/or Component** | **Material** | **Environment** | **Aging Effect/Mechanism** | **Aging Management Program (AMP)/TLAA** | **Further Evaluation** |
| ~~M~~D | IV.B4.RP-254a | ~~3.1-1, 051b~~ | ~~Lower grid assembly: alloy X-750 lower grid shock pad bolt locking devices (Three Mile Island Unit 1 only)~~ | ~~Nickel alloy~~ | ~~Reactor coolant and neutron flux~~ | ~~Cracking due to fatigue~~ | ~~AMP XI.M16A, "PWR Vessel Internals"~~ | ~~Yes~~ |
| ~~M~~D | IV.B4.RP-254 | ~~3.1-1, 051b~~ | ~~Lower grid assembly: alloy X-750 lower grid shock pad bolts (Three Mile Island Unit 1 only)~~ | ~~Nickel alloy~~ | ~~Reactor coolant and neutron flux~~ | ~~Cracking due to SCC~~ | ~~AMP XI.M16A, "PWR Vessel Internals," and AMP XI.M2, "Water Chemistry"~~ | ~~Yes~~ |
| M | IV.B4.RP-246a | 3.1-1, 051b | Lower grid assembly: ~~upper thermal shield (UTS) bolt locking devices and~~ lower thermal shield (LTS) bolt locking devices | Stainless steel, nickel alloy | Reactor coolant and neutron flux | Cracking due to fatigue | AMP XI.M16A, "PWR Vessel Internals" | Yes |
| M | IV.B4.RP-246b | 3.1-1, 058b | Lower grid assembly: ~~upper thermal shield (UTS) bolt locking devices and~~ lower thermal shield (LTS) bolt locking devices | Stainless steel, nickel alloy | Reactor coolant and neutron flux | Loss of material due to wear; changes in dimensions due to void swelling or distortion | AMP XI.M16A, "PWR Vessel Internals" | Yes |
| M | IV.B4.RP-246 | 3.1-1, 051b | Lower grid assembly: ~~upper thermal shield (UTS) bolts and~~ lower thermal shield (LTS) bolts | Stainless steel, nickel alloy | Reactor coolant and neutron flux | Cracking due to SCC | AMP XI.M16A, "PWR Vessel Internals," and AMP XI.M2, "Water Chemistry" | Yes |
| N | IV.B4.RP-246c | 3.1-1, 051b | Core barrel assembly: upper thermal shield (UTS) bolts | Stainless steel; nickel alloy | Reactor coolant and neutron flux | Cracking due to SCC | AMP XI.M16A, "PWR Vessel Internals," and AMP XI.M2, “Water Chemistry” | Yes |

| IV<br>Table B4 | REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM<br>Reactor Vessel Internals (PWR)—Babcock & Wilcox | | | | | | | |
|---|---|---|---|---|---|---|---|---|
| **New, Modified, Deleted, Edited Item** | **Item** | **SRP Item (Table, ID)** | **Structure and/or Component** | **Material** | **Environment** | **Aging Effect/Mechanism** | **Aging Management Program (AMP)/TLAA** | **Further Evaluation** |
| N | IV.B4.RP-246d | 3.1-1, 051b | Core barrel assembly: upper thermal shield (UTS) bolt locking devices | Stainless steel; nickel alloy | Reactor coolant and neutron flux | Cracking due to fatigue | AMP XI.M16A, "PWR Vessel Internals" | Yes |
| N | IV.B4.RP-246e | 3.1-1, 058b | Core barrel assembly: upper thermal shield (UTS) bolt locking devices | Stainless steel; nickel alloy | Reactor coolant and neutron flux | Loss of material due to wear; changes in dimension due to void swelling or distortion | AMP XI.M16A, "PWR Vessel Internals" | Yes |
| M | IV.B4.RP-260 | 3.1-1, 058b | Lower grid ~~fuel~~ assembly: ~~(a)~~ ~~accessible~~ pads~~; (b) accessible~~, pad-to-rib section welds~~; (c) accessible alloy X-750~~, dowels, cap screws and their locking devices | Stainless steel, nickel alloy | Reactor coolant and neutron flux | Loss of fracture toughness due to neutron irradiation embrittlement | AMP XI.M16A, "PWR Vessel Internals" | Yes |
| M | IV.B4.RP-260a | 3.1-1, 051b | Lower grid ~~fuel~~ assembly: ~~(a)~~ pads~~; (b)~~, pad-to-rib section welds~~; (c) alloy X-750~~, dowels, cap screws and their locking devices | Stainless steel, nickel alloy | Reactor coolant and neutron flux | Cracking due to SCC or fatigue | AMP XI.M16A, "PWR Vessel Internals," and AMP XI.M2, "Water Chemistry" (for SCC mechanisms only) | Yes |
| M | IV.B4.RP-251a | 3.1-1, 058a | Plenum cover assembly: plenum cover weldment rib pads ~~and~~, plenum cover support flange, plenum cover support ring | Stainless steel | Reactor coolant and neutron flux | Loss of material due to wear; loss of preload (wear) | AMP XI.M16A, "PWR Vessel Internals" | Yes |

| **IV<br>Table B4** | **REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM<br>Reactor Vessel Internals (PWR)—Babcock & Wilcox** | | | | | | | |
|---|---|---|---|---|---|---|---|---|
| **New, Modified, Deleted, Edited Item** | **Item** | **SRP Item (Table, ID)** | **Structure and/or Component** | **Material** | **Environment** | **Aging Effect/Mechanism** | **Aging Management Program (AMP)/TLAA** | **Further Evaluation** |
| N | IV.B4.R-423 | 3.1-1, 118 | Reactor vessel internal components<u>, or reactor vessel internal component-specific basis for a specified RVI component</u> | Stainless steel, nickel alloy | Reactor coolant, neutron flux | Cracking due to SCC, ~~irradiation-assisted SCC~~<u>IASCC</u>, cyclic loading, fatigue | ~~Plant-specific aging management program~~<u>Plant-specific aging management program, or AMP XI.M16A, "PWR Vessel Internals," and AMP XI.M2, "Water Chemistry" (SCC and IASCC only), for cases where a specified component is subject to a site-specific or component-specific aging management basis</u> | Yes |

| IV Table B4 | REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM Reactor Vessel Internals (PWR)—Babcock & Wilcox | | | | | | | |
|---|---|---|---|---|---|---|---|---|
| **New, Modified, Deleted, Edited Item** | **Item** | **SRP Item (Table, ID)** | **Structure and/or Component** | **Material** | **Environment** | **Aging Effect/Mechanism** | **Aging Management Program (AMP)/TLAA** | **Further Evaluation** |
| N | IV.B4.R-424 | 3.1-1, 119 | Reactor vessel internal components, or reactor vessel internal component-specific basis for a specified RVI component | Stainless steel, nickel alloy | Reactor coolant, neutron flux | Loss of fracture toughness due to neutron irradiation embrittlement or thermal aging embrittlement; changes in dimensions due to void swelling or distortion; loss of preload due to thermal and irradiation-enhanced stress relaxation or creep; loss of material due to wear | Plant-specific aging management program, or AMP XI.M16A, "PWR Vessel Internals," for cases where a specified component is subject to a site-specific or component-specific aging management basis | Yes |
| ~~M~~D | IV.B4.RP-382 | ~~3.1-1, 032~~ | ~~Reactor vessel internals: ASME Section XI, Examination Category B-N-3 core support structure components (not already identified as "Existing Programs" components in MRP-227-A)~~ | ~~Stainless steel, nickel alloy, cast austenitic stainless steel~~ | ~~Reactor coolant and neutron flux~~ | ~~Cracking due to fatigue, SCC, or irradiation-assisted SCC; loss of material due to wear~~ | ~~AMP XI.M1, "ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD"~~ | ~~No~~ |
| M | IV.B4.RP-352 | 3.1-1, 051b | Upper grid assembly: ~~alloy X-750~~ dowel-to-upper grid fuel assembly support pad welds (all plants except Davis-Besse) | Nickel alloy | Reactor coolant and neutron flux | Cracking due to SCC | AMP XI.M16A, "PWR Vessel Internals," and AMP XI.M2, "Water Chemistry" (for SCC mechanisms only) | Yes |

| IV<br>Table B4 | REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM<br>Reactor Vessel Internals (PWR)—Babcock & Wilcox | | | | | | | |
|---|---|---|---|---|---|---|---|---|
| **New, Modified, Deleted, Edited Item** | **Item** | **SRP Item (Table, ID)** | **Structure and/or Component** | **Material** | **Environment** | **Aging Effect/Mechanism** | **Aging Management Program (AMP)/TLAA** | **Further Evaluation** |
| N | IV.B4.RP-386 | 3.1-1, 051b | Lower Grid Assembly: lower grid rib section | Stainless steel | Reactor coolant and neutron flux | Cracking due to SCC or fatigue | AMP XI.M16A, "PWR Vessel Internals," and AMP XI.M2, "Water Chemistry" (for SCC mechanisms only) | Yes |
| N | IV.B4.RP-386a | 3.1-1, 058b | Lower Grid Assembly: lower grid rib section | Stainless steel | Reactor coolant and neutron flux | Loss of fracture toughness due to neutron irradiation embrittlement | AMP XI.M16A, "PWR Vessel Internals," and AMP XI.M2, "Water Chemistry" (for SCC mechanisms only) | Yes |

## APPENDIX B.4

## PROPOSED REVISIONS TO GALL-SLR REPORT TABLE IV.E, “COMMON MISCELLANEOUS MATERIAL/ENVIRONMENT COMBINATIONS”

NUREG-2191, Volumes 1 and 2, “Generic Aging Lessons Learned for Subsequent License Renewal (GALL-SLR) Report,” Table IV.E, “Common Miscellaneous Material/Environment Combinations,” addresses miscellaneous material/environment combinations that may be found throughout the reactor vessel, internals, and reactor coolant systems, structures, and components.

Proposed revisions to Table IV.E of the GALL-SLR Report are provided in redline format.  This AMR item supersedes the respective item in GALL-SLR Report, Revision 0, Table IV.E.

GALL-SLR Report Table IV.E Proposed Revisions

| IV Table E | REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM Common Miscellaneous Material/Environment Combinations | | | | | | | |
|---|---|---|---|---|---|---|---|---|
| **New, Modified, Deleted, Edited Item** | **Item** | **SRP Item (Table, ID)** | **Structure and/or Component** | **Material** | **Environment** | **Aging Effect/Mechanism** | **Aging Management Program (AMP)/TLAA** | **Further Evaluation** |
| N | IV.E.R-444 | 3.1-1, 114 | Reactor coolant system components: Components defined as ASME Section XI components (e.g., ASME Code Class 1 reactor coolant pressure boundary components, reactor interior attachments, or core support structure components, ASME Class 2 or 3 components, including associated pressure-retaining welds) not managed by other AMR line items in GALL-SLR Chapter IV | Any | Applicable internal or external environment | Cracking due to SCC, IGSCC (stainless steel or nickel alloy components only), cyclic loading; loss of material due to general corrosion (steel only), pitting corrosion, crevice corrosion, wear | AMP XI.M1, "ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD," and AMP XI.M2, "Water Chemistry" (water chemistry-related or corrosion-related aging effect mechanisms only) | No |

## APPENDIX C

## PROPOSED REVISIONS TO SRP-SLR SECTION 3.1.2.2.9 (AMR FURTHER EVALUATION ACCEPTANCE CRITERIA) AND SRP-SLR SECTION 3.1.3.2.9 (AMR FURTHER EVALUATION REVIEW PROCEDURES)

NUREG-2192, "Standard Review Plan for Review of Subsequent License Renewal Applications for Nuclear Power Plants" (SRP-SLR), Sections 3.1.2.2.9 and 3.1.3.2.9, provide staff guidance for the acceptance criteria and review procedures, respectively, for the further evaluation item related to aging management of pressurized-water reactor vessel internals. These sections are reproduced below in their entirety with revisions provided in redline format, and supersede SRP-SLR, Revision 0, Sections 3.1.2.2.9 and 3.1.3.2.9.

SRP-SLR Further Evaluation Proposed Revisions

*3.1.2.2.9 Aging Management of Pressurized Water Reactor Vessel Internals (Applicable to Subsequent License Renewal Periods Only)*

Electric Power Research Institute (EPRI) Topical Report (TR)-1022863, "Materials Reliability Program: Pressurized Water Reactor Internals Inspection and Evaluation Guidelines (MRP-227-A)" (Agencywide Documents Access and Management System (ADAMS) Accession Nos. ML12017A191 through ML12017A197 and ML12017A199), ~~provides~~ provided the industry's ~~current aging management~~initial set of aging management inspection and evaluation (I&E) recommendations for the reactor vessel internal (RVI) components that are included in the design of a PWR facility. Since the issuance of MRP-227-A on January 9, 2012, EPRI updated its I&E guidelines for the PWR RVI components in Topical Report No. 3002017168, "Materials Reliability Program: Pressurized Water Reactor Internals Inspection and Evaluation Guidelines (MRP-227, Revision 1-A)" (ADAMS Accession No. ML19339G350). MRP-227, Revision 1-A, incorporated the industry's bases for resolving operating experience and industry lessons learned resulting from component-specific inspections performed since the issuance of MRP-227-A in January 2012. The staff found the guidelines in MRP-227, Revision 1-A, acceptable, as documented in a staff-issued safety evaluation dated April 25, 2019 (ADAMS Accession No. ML19081A001) and approved the topical report for use as documented in the staff's letters to the EPRI Materials Reliability Program (MRP) dated February 19, 2020 and July 7, 2020 (ADAMS Accession Nos. ML20006D152 and ML20175A149).

In ~~this report~~ MRP-227, Revision 1-A, the EPRI ~~Materials Reliability Program (~~MRP~~)~~ identified that the following aging mechanisms may be applicable to the design of the RVI components in these types of facilities: (a) stress corrosion cracking (SCC), (b) irradiation-assisted stress corrosion cracking (IASCC), (c) fatigue, (d) wear, (e) neutron irradiation embrittlement, (f) thermal aging embrittlement, (g) void swelling and irradiation growth or component distortion, or (h) thermal or irradiation-enhanced stress relaxation or irradiation enhanced creep. ~~The methodology in MRP-227-A was approved by the NRC in a safety evaluation dated December 16, 2011 (ADAMS Accession No. ML11308A770), which includes those plant-specific applicant/licensee action items that a licensee or applicant applying the MRP-227-A report would need to address and resolve and apply to its licensing basis.~~

The EPRI MRP's functionality analysis and failure modes, effects, and criticality analysis bases for grouping Westinghouse-designed, B&W-designed and Combustion Engineering (CE)-designed RVI components into ~~these~~ the applicable inspection categories (as evaluated in MRP-227, Revision 1-A) ~~was~~ were based on an assessment of aging effects and relevant time-dependent aging parameters through a cumulative 60-year licensing period (i.e., 40 years for the initial operating license period plus an additional 20 years during the initial period of extended operation). The EPRI MRP's ~~has not assessed~~assessment in MRP-227, Revision 1-A, did not evaluate whether operation of Westinghouse-designed, B&W-designed and CE-designed reactors during an SLR operating period (60 to 80 years) would have any impact on the existing susceptibility rankings and inspection categorizations for the RVI components in these designs, as defined in MRP-227, Revision 1-A or ~~its~~ the applicable MRP background documents (e.g., MRP-191, Revision 1, for Westinghouse-designed or CE-designed RVI components or MRP-189, Revision 2, for B&W-designed components).

As described in GALL-SLR Report AMP XI.M16A, the applicant may use the MRP-227, Revision 1-A based AMP as an initial reference basis for developing and defining the AMP that will be applied to the RVI components for the subsequent period of extended operation. However, to use this alternative basis, GALL-SLR Report AMP XI.M16A recommends that the MRP-227, Revision 1-A based AMP be enhanced to include a gap analysis of the components that are within the scope of the AMP. The gap analysis is a basis for identifying and justifying ~~any potential~~ changes to the MRP-227, Revision 1-A based program that ~~may be~~are necessary to provide reasonable assurance that the effects of age-related degradation will be managed during the subsequent period of extended operation. The criteria for the gap analysis are described in GALL-SLR Report AMP XI.M16A. If a gap analysis is needed to establish the appropriate aging management criteria for the RVI components, the applicant has the option of including the gap analysis in the SLRA for its reactor unit(s) or making the gap analysis and any supporting gap analysis documents available in the in-office audit portal for the SLRA review.

Subsequent license renewal (SLR) applicants for units of a PWR design will no longer need to include separate SLRA Appendix C section responses in resolution of the A/LAIs previously issued on MRP-227-A because the A/LAIs were resolved and closed by the staff in the April 25, 2019, safety evaluation for MRP-227, Revision 1-A. The sole A/LAI issued by the staff in the safety evaluation dated April 25, 2019, relates to an applicant's methods and timing of inspections that will be applied to the baffle-to-former bolts or core shroud bolts in the plant design. Since an applicant's resolution of this A/LAI can be appropriately addressed in the "Operating Experience" program element discussion for the AMP and in the applicant's basis document for the AMP, a separate SLRA Appendix C response for the A/LAI is unnecessary.

Alternatively, the PWR SLRA may define a plant-specific AMP for the RVI components to demonstrate that the RVI components will be managed in accordance with the requirements of 10 CFR 54.21(a)(3) during the proposed subsequent period of extended operation. Components to be inspected, parameters monitored, monitoring methods, inspection sample size, frequencies, expansion criteria, and acceptance criteria are justified in the SLRA. ~~The~~ If the AMP is a plant-specific program, the NRC staff will assess the adequacy of the plant-specific AMP against the criteria for the 10 AMP program elements that are defined in Section A.1.2.3 of SRP-SLR Appendix A.1.

*3.1.3.2.9 Aging Management of Pressurized Water Reactor Vessel Internals (Applicable to Subsequent License Renewal Periods Only)*

EPRI ~~TR–1022863~~Topical Report No. 3002017168, "Materials Reliability Program: Pressurized Water Reactor Internals Inspection and Evaluation Guidelines (MRP-227, Revision 1-A)" (ADAMS Accession No~~s~~. ML19339G350~~ML12017A191 through ML12017A197 and ML12017A199~~), provides the industry's ~~current~~ updated aging management recommendations for the RVI components that are included in the design of a PWR facility, based on an analysis of plant operation for 60 years. The review procedures in this section are based on the staff’s assumption that a PWR SLR applicant’s PWR vessel internals AMP will be based on the I&E guidelines in MRP-227, Revision 1-A for the AMP that will be applied and implemented during the subsequent period of extended operation. The rationale for this assumption is based on the MRP-defined “Needed Requirement” in Section 7.3 of MRP-227, Revision 1-A, which states that the update of MRP-based program “shall be implemented by January 1, 2022.”

In ~~this report~~MRP-227, Revision 1-A, the EPRI MRP identified that the following aging mechanisms may be applicable to the design of the RVI components in these types of facilities: (a) stress corrosion cracking (SCC), (b) irradiation-assisted stress corrosion cracking (IASCC), (c) fatigue, (d) wear, (e) neutron irradiation embrittlement, (f) thermal aging embrittlement, (g) void swelling and irradiation growth or distortion, or (h) thermal or irradiation-enhanced stress relaxation or irradiation enhanced creep. ~~The methodology in~~ The staff approved MRP-227, Revision 1-A ~~was approved by the NRC~~ in a safety evaluation dated ~~December 16, 2011~~April 25, 2019 (ADAMS Accession No. ~~ML11308A770~~ML19081A001).~~, which includes~~ In that safety evaluation, the staff resolved and closed all ~~those plant-specific~~ applicant/licensee action items (A/LAIs) that were previously issued on the previous version of the I&E guidelines (i.e., ~~a licensee or applicant applying~~those in the MRP-227-A report). The assessments of RVI components in ~~the~~ MRP-227, Revision 1-A, ~~report~~ and the MRP-defined background reports for MRP-227, Revision 1-A have not been updated based on an assessment of aging effects over an 80-year operating period.

If a plant-specific AMP is proposed for the RVI components, the reviewer evaluates the adequacy of the applicant’s AMP on a case-by-case basis against the criteria for plant-specific AMP program elements defined in Sections A.1.2.3.1 through A.1.2.3.10 of SRP-SLR Appendix A.1. The reviewer verifies that the applicant has defined both the type of performance monitoring, condition monitoring, preventative monitoring, or mitigative monitoring ~~AMP~~ activities that will be used for aging management of the RVI components and the specific program element criteria for the AMP that will be used to manage age-related effects in the RVI components during the subsequent period of extended operation.

If a PWR applicant for SLR proposes to use GALL-SLR Report AMP XI.M16A, “PWR Vessel Internals,” as the basis for aging management, the staff reviews the program elements of the AMP against the program element criteria defined in AMP XI.M16A. The staff verifies that the applicant has addressed the relevancy of the A/LAI for MRP-227, Revision 1-A in the “Operating Experience” program element of the AMP, or in the applicant’s technical basis document or procedure for the AMP. The staff also verifies that the proposed program includes a gap analysis that provides the identification and justification of:

- Components that screen in for additional aging effects or mechanisms when assessed for aging through the end of the subsequent period of extended operation

- Components that previously screened in for an aging effect or mechanism and the severity of that aging effect or mechanism could significantly increase during the subsequent period of extended operation

- Changes to the existing MRP-227, Revision 1-A program characteristics or criteria, including, but not limited to, changes in inspection categories, inspection criteria, or primary-to-expansion component criteria and relationships

~~The~~ If a gap analysis is needed to establish the appropriate aging management criteria for the RVI components, the staff evaluates the adequacy and justification of the gap analysis in the safety evaluation report for the SLRA. ~~, s~~Specifically, the staff's review should focus on the following aspects of the gap analysis:

- The gap analysis methodology
- The components that screened in for additional aging effects or mechanisms when assessed for aging through the end of the subsequent period of extended operation
- The components for which a previously screened in aging effect or mechanism has been identified as potentially more severe during the subsequent period of extended operation
- Components whose AMP inspection categories have changed from those previously identified for the components in MRP-227, Revision 1-A
- Proposed changes to the aging management program characteristics or criteria identified in the SLRA

For those RVI components that screened in for additional aging effects or mechanisms, or that are subject to site-specific or component-specific changes in the EPRI MRP's I&E protocols for the components, the staff also confirms that the applicant has included and justified appropriate AMR line items for the components. The applicant may use the updated version of GALL-SLR Report Item IV.B2.R-423, IV.B3.R-423, or IV.B4.R-423 to address any RVI component for which the EPRI MRP I&E protocols for managing cracking or specific cracking mechanisms in the component are being updated or adjusted on a site-specific or component-specific basis. The applicant may use the updated version of GALL-SLR Report Items IV.B2.R-424, IV.B3.R-424, or IV.B4.R-424 to address any RVI component for which the EPRI MRP I&E protocols for managing non-cracking effects or mechanisms in the component are being updated or adjusted on a site-specific or component-specific basis.

Otherwise an applicant may use an NRC-approved generic ~~methodology~~report such as an approved revision of MRP-227 that considers an operating period of 80 years. In this case, the staff reviews any responses to action items on the aging management methods that may be identified in the NRC approval of the generic ~~methodology~~report.

**APPENDIX D**

**PROPOSED REVISIONS TO GALL-SLR REPORT AMP XI.M16A, "PWR VESSEL INTERNALS," AND RELATED FSAR SUPPLEMENT EXAMPLE IN GALL-SLR REPORT TABLE XI-01**

NUREG-2191, Volumes 1 and 2, "Generic Aging Lessons Learned for Subsequent License Renewal (GALL-SLR) Report," Aging Management Program (AMP) XI.M16A, "PWR Vessel Internals," describes one acceptable way to manage aging effects related to pressurized-water reactor (PWR) vessel internals for subsequent license renewal. This AMP is reproduced below in its entirety, with revisions provided in redline format. It supersedes GALL-SLR Report, Revision 0, AMP XI.M16A.

This appendix also provides a redline version of the AMP XI.M16A final safety analysis report (FSAR) supplement summary in GALL-SLR Report Table XI-01, "FSAR Supplement Summaries for GALL-SLR Report Chapter XI Aging Management Programs." This entry modifies GALL-SLR Report, Revision 0, Table XI-01.

GALL-SLR Report Aging Management Program XI.M16A Proposed Revisions

## XI.M16A PWR VESSEL INTERNALS

### Program Description

This program is used to manage the effects of age-related degradation mechanisms that are applicable to the pressurized water reactor (PWR) reactor vessel internal (RVI) components. These aging effects include: (a) cracking, including stress corrosion cracking (SCC), primary water stress corrosion cracking (PWSCC), irradiation-assisted stress corrosion cracking (IASCC), and cracking due to fatigue/cyclic loading; (b) loss of material induced by wear; (c) loss of fracture toughness due to thermal aging and neutron irradiation embrittlement; (d) changes in dimensions due to void swelling or distortion; and (e) loss of preload due to thermal and irradiation-enhanced stress relaxation or creep.

In the absence of an acceptable generic ~~methodology~~ report such as an approved revision of Materials Reliability Program (MRP)-227 that considers an operating period of 80 years, this program may be based on an existing plant program that is consistent with Electric Power Research Institute (EPRI) ~~Technical~~ Topical Report No. 3002017168~~1022863~~, "Materials Reliability Program: Pressurized Water Reactor ~~(PWR)~~ Internals Inspection and Evaluation Guidelines~~,"~~ (MRP-227, Revision 1-A)," which is implemented in accordance with Nuclear Energy Institute (NEI) 03-08, "Guideline for the Management of Materials Issues." The staff ~~approved~~ found the ~~augmented~~ updated ~~inspection and evaluation (~~I&E~~)~~ guidelines and criteria for PWR RVI components acceptable, as documented in the staff's safety evaluation of April 25, 2019 (ADAMS Accession No. ML19081A001), and approved the use of MRP-227, Revision 1-A, for PWR-specific design bases in the staff's letters to the EPRI MRP dated February 19, 2020 and July 7, 2020 (ADAMS Accession Nos. ML20006D152 and ML20175A149)~~NRC Safety Evaluation (SE), Revision 1, on MRP-227 by letter dated December 16, 2011~~.

Because the guidelines of MRP-227, Revision 1-A, are based on an analysis of the RVI that considers the operating conditions up to a 60-year operating period, these guidelines are supplemented through a gap analysis that identifies enhancements to the program that are needed to address an 80-year operating period. In this program, the term “MRP-227~~-A~~ (as supplemented)” is used to describe either MRP-227, Revision 1-A, as supplemented by this gap analysis, or an acceptable generic ~~methodology~~ report such as an approved revision of MRP-227 that considers an operating period of 80 years.

The program applies the guidance in MRP-227~~-A~~ (as supplemented) for inspecting, evaluating, and, if applicable, dispositioning non-conforming RVI components at the facility. 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 identified exceeds the expected levels.

The methodology used in the development of MRP-227, Revision 1-A, ~~guidance~~ for selecting RVI components for inclusion in the inspection sample is based on a four-step ranking process. Through this process, the RVIs for ~~all three~~Westinghouse and Combustion Engineering (CE) PWR designs were assigned to one of the following four ~~groups~~inspection categories: “Primary,” “Expansion,” “Existing Programs,” ~~and~~ or “No Additional Measures.” Through this process, the RVIs for Babcock & Wilcox (B&W) PWR designs were assigned to one of the following three inspection categories: “Primary,” “Expansion,” or “No Additional Measures.” Definitions of each ~~group~~ category are provided in MRP-227, Revision 1-A.

~~In the absence of an acceptable generic methodology such as an approved revision of MRP-227 that considers an operating period of 80 years, the gap analysis described below is used to provide reasonable assurance that the aging management for the RVI components identified in the four groups is appropriate for 80 years of operation.~~

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 inspected because they are expected to show the leading indications of the degradation effects.~~, with another set~~ The category of “Expansion” internals component locations ~~that are~~ is specified to expand the sample should the indications from the “Primary” components be more severe than anticipated.

The degradation effects in a third set of internals locations (which apply only to the RVI components in Westinghouse- or CE-designed PWRs) are deemed to be adequately managed by “Existing Programs,” such as American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code), Section XI,[1] Examination Category B-N-3, examinations of core support structures. A fourth set of internals locations are deemed to require “No Additional Measures.”

In the absence of an acceptable generic report such as an approved revision of MRP-227 that considers an operating period of 80 years, the gap analysis described below is used to provide reasonable assurance that the aging management activities designated for the RVI components identified in the four groups is appropriate for 80 years of operation. The gap analysis may include and incorporate supplemental guidelines developed and recommended for the RVI components.

If the program is based on MRP-227, Revision 1-A, with a gap analysis, the inspection categories, inspection criteria, and other program characteristics ~~required by~~established in

[1] GALL-SLR Report Chapter I, Table 1, identifies the ASME Code Section XI editions and addenda that are acceptable to use for this AMP.

MRP-227, Revision 1-A, are identified and justified for each component in the applicable program elements. The justification should focus on the aging management of ~~the~~ any additional aging considerations (i.e., new aging effect/mechanism) during the subsequent period of extended operation. The acceptance criteria in the Standard Review Plan for Review of Subsequent License Renewal Applications for Nuclear Power Plants (SRP-SLR), Section 3.1.2.2.9 and the review procedures in Section 3.1.3.2.9 provide additional information.

## Evaluation and Technical Basis

1. ***Scope of Program***: The scope of the program includes all RVI components based on the plant's applicable nuclear steam supply system design. The scope of the program applies the ~~methodology and guidance~~guidelines in MRP-227~~-A~~ (as supplemented), which provides ~~an~~ augmented inspection and flaw evaluation ~~methodology~~ guidelines 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. Since these types of AMPs are considered to be "living" programs by the licensees implementing the programs, the scope of program may also include additional reports, documents or guidelines recommended for implementation by the EPRI MRP, PWR Owners Group, or industry vendors. This may include (but is not limited to) applicable WCAP or BAW technical/topical reports issued by Westinghouse or B&W, or supplemental guidelines or industry alert letters issued by the EPRI MRP, PWR Owners Group, or industry vendors.

   The scope of components includes core support structures, those RVI components that serve an intended license renewal safety function pursuant to criteria in Title 10 of the *Code of Federal Regulations* (10 CFR) 54.4(a)(1), and other RVI components whose failure could prevent satisfactory accomplishment of any of the functions identified in 10 CFR 54.4(a)(1)(i), (ii), or (iii). In addition, ASME Code, Section XI includes inspection requirements for PWR removable core support structures in Table IWB-2500-1, Examination Category B-N-3, which are in addition to any inspections that are implemented in accordance with MRP-227~~-A~~ (as supplemented).

   The scope of the program does not include consumable items, such as fuel assemblies, reactivity control assemblies, and nuclear instrumentation. 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 managed in accordance with an applicant's AMP that corresponds to GALL-SLR Report AMP XI.M1, "ASME Code, Section XI Inservice Inspection, Subsections IWB, IWC, and IWD."

   This program element specifies ~~if~~ whether the program is based on an existing program that is consistent with MRP-227, Revision 1-A, with a gap analysis, or ~~if it~~the program is based on an acceptable generic ~~methodology~~ report that covers an 80-year service life for the RVI components, such as an approved revision of MRP-227 that considers an operating period of 80 years. If based on MRP-227, Revision 1-A, with a gap analysis, the scope of the program focuses on identification and justification of the following:

   a. Components that screen in for additional aging effects or mechanisms when assessed for the 60–80 year operating period.

b. Components that previously screened in for an aging effect or mechanism and the severity of that aging effect or mechanism could significantly increase for the 60–80 year operating period.

c. Changes to the existing MRP-227, Revision 1-A, program characteristics or criteria, including but not limited to changes in inspection categories, inspection criteria, or primary-to-expansion component criteria and relationships.

2. ***Preventive Actions***: The program relies on PWR water chemistry control to prevent or mitigate 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)]. Reactor coolant water chemistry is monitored and maintained in accordance with the Water Chemistry Program, as described in GALL-SLR Report AMP XI.M2, “Water Chemistry.”

3. ***Parameters Monitored or Inspected***: The program manages the following age-related degradation effects and mechanisms that are applicable in general to RVI components at the facility: (a) cracking induced by SCC, PWSCC, IASCC, or fatigue/cyclic loading; (b) loss of material induced by wear; (c) loss of fracture toughness induced by thermal aging and neutron irradiation embrittlement; (d) changes in dimensions due to void swelling or distortion; and (e) loss of preload due to 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-destructive examination (NDE) method or for relevant flaw presentation signals if a volumetric ultrasonic testing (UT) method is used as the NDE method. For the management of loss of material, the program monitors for gross or abnormal surface 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 thermal aging or neutron irradiation embrittlement. Instead, the impact of loss of fracture toughness on component integrity is indirectly managed by: (1) using visual or volumetric examination techniques to monitor for cracking in the components, and (2) applying applicable reduced fracture toughness properties in the flaw evaluations, in cases where cracking is detected in the components and is extensive enough to necessitate a supplemental flaw growth or flaw tolerance evaluation. The program uses physical measurements to monitor for any dimensional changes due to void swelling or distortion.

   Specifically, the program implements the parameters monitored/inspected criteria consistent with the applicable tables in Section 4, “Aging Management Requirements,” in MRP-227~~-A~~ (as supplemented).

4. ***Detection of Aging Effects***: The inspection methods are defined and established in Section 4 of MRP-227~~-A~~ (as supplemented). Standards for implementing the inspection methods are defined and established in MRP-228. In all cases, well-established inspection methods are 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. Surface examinations may also be used as an

alternative to visual examinations for 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). VT-3 visual methods may be applied for the detection of cracking in non-redundant RVI components only when the flaw tolerance of the component, as evaluated for reduced fracture toughness properties, is known and the component has been shown to be tolerant of easily detected large flaws, even under reduced fracture toughness conditions. VT-3 visual methods are acceptable for the detection of cracking in redundant RVI components (e.g., redundant bolts or pins used to secure a fastened RVI assembly).

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.

The program adopts the guidance in MRP-227~~-A~~ (as supplemented) for defining the "Expansion Criteria" that need to be applied to the inspection findings of "Primary" components and for expanding the examinations to include additional "Expansion" components. RVI component inspections are performed consistent with the inspection frequency and sampling bases for "Primary" components, "Existing Programs" components, and "Expansion" components in MRP-227~~-A~~ (as supplemented).

In some cases (as defined in MRP-227, Revision 1-A), 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 dimensions due to void swelling or distortion.

Inspection coverages for "Primary" and "Expansion" RVI components are implemented consistent with ~~Sections 3.3.1 and 3.3.2 of the NRC SE, Revision 1, on~~those established in MRP-227, Revision 1-A, or as modified by a gap analysis.

This program element should justify the appropriateness of the inspection methods, sample size criteria, and inspection frequency criteria for managing the effects of degradation during the subsequent period of extended operation, including any changes to these criteria from their ~~prior~~ assessment in MRP-227, Revision 1-A.

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~~-A~~ (as supplemented) and its subsections. Component reinspection frequencies for "Primary" and "Expansion" category components are defined in specific tables in Section 4 of the MRP-227~~-A~~ report (as supplemented). The examination and re-examinations that are implemented in accordance with MRP-227~~-A~~ (as supplemented), together with the criteria specified in MRP-228 for inspection ~~methodologies~~standards, inspection procedures, and inspection personnel, provide for timely detection, reporting, and implementation of corrective actions for the aging effects and mechanisms managed by the program.

The program applies applicable fracture toughness properties, including reductions for thermal aging or neutron embrittlement, in the flaw evaluations of the components in cases where cracking is detected in an RVI component and is extensive enough to warrant a supplemental flaw growth or flaw tolerance evaluation.

For singly-represented components, the program includes criteria to evaluate the aging effects in the inaccessible portions of the components and the resulting impact on the intended function(s) of the components. For redundant components (such as redundant bolts, screws, pins, keys, or fasteners, some of which are accessible to inspection and some of which are not accessible to inspection), the program includes criteria to evaluate the aging effects in the population of components that are inaccessible by the applicable inspection technique and the resulting impact on the intended function(s) of the assembly containing the components.

Flaw evaluation methods, including recommendations for flaw depth sizing and for crack growth determinations as well as for performing applicable limit load, linear elastic and elastic-plastic fracture analyses of relevant flaw indications, are defined in MRP-227-A (as supplemented).

6. ***Acceptance Criteria****:* Section 5 of MRP-227~~-A~~ (as supplemented), which includes Table 5-1 for B&W-designed RVIs, Table 5-2 for CE-designed RVIs, and Table 5-3 for Westinghouse-designed RVIs, provides the specific examination and flaw evaluation acceptance criteria for the “Primary” and “Expansion” RVI component examination methods. Consistent with the criteria in MRP-227, Revision 1-A, the acceptance criteria for some “Expansion” category components may be established through performance of a component-specific analysis or component replacements, particularly if the components are inaccessible for inspection or the industry has yet to develop adequate inspection methods for the components. For RVI components addressed by examinations performed in accordance with the ASME Code, Section XI, the acceptance criteria in IWB-3500 are applicable. For RVI components covered by other “Existing Programs,” the acceptance criteria are described within the applicable reference document. As applicable, the program establishes acceptance criteria for any physical measurement monitoring methods that are credited for aging management of particular RVI components.

This program element should justify the appropriateness of the acceptance criteria for managing the effects of degradation during the subsequent period of extended operation, including any changes to acceptance criteria based on the gap analysis.

7. ***Corrective Actions***: Results that do not meet the acceptance criteria are addressed in the applicant’s corrective action program under those specific portions of the quality assurance (QA) program that are used to meet Criterion XVI, “Corrective Action,” of 10 CFR Part 50, Appendix B. Appendix A of the Generic Aging Lessons Learned for Subsequent License Renewal (GALL-SLR) Report describes how an applicant may apply its 10 CFR Part 50, Appendix B, QA program to fulfill the corrective actions element of this AMP for both safety-related and nonsafety-related structures and components (SCs) within the scope of this 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.

The implementation of the guidance in MRP-227~~-A~~ (as supplemented), 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 or its equivalent, as applicable.

Other alternative corrective actions bases may be used to disposition relevant conditions if they have been previously approved or endorsed by the NRC. Alternative corrective actions not approved or endorsed by the NRC will be submitted for NRC approval prior to their implementation.

8. ***Confirmation Process***: The confirmation process is addressed through those specific portions of the QA program that are used to meet Criterion XVI, "Corrective Action," of 10 CFR Part 50, Appendix B. Appendix A of the GALL-SLR Report describes how an applicant may apply its 10 CFR Part 50, Appendix B, QA program to fulfill the confirmation process element of this AMP for both safety-related and nonsafety-related SCs within the scope of this program.

   Site quality assurance procedures, review and approval processes, and administrative controls are implemented in accordance with the recommendations of NEI 03-08 and the requirements of 10 CFR Part 50, Appendix B, or their equivalent, as applicable. The implementation of the guidance in Section 7 of MRP-227~~-A~~ (as supplemented), in conjunction with NEI 03-08 and other guidance documents, reports, or ~~methodologies~~ guidelines referenced in this AMP, provides an acceptable level of quality and an acceptable basis for confirming the quality of inspections, flaw evaluations, and corrective actions.

9. ***Administrative Controls****:* Administrative controls are addressed through the QA program that is used to meet the requirements of 10 CFR Part 50, Appendix B, associated with managing the effects of aging. Appendix A of the GALL-SLR Report describes how an applicant may apply its 10 CFR Part 50, Appendix B, QA program to fulfill the administrative controls element of this AMP for both safety-related and nonsafety-related SCs within the scope of this program.

   The administrative controls for these types of programs, including their implementing procedures and review and approval processes, are implemented in accordance with the recommended industry guidelines and criteria in NEI 03-08, and are under existing site 10 CFR 50 Appendix B, Quality Assurance Programs, or their equivalent, as applicable. The basis defined in Section 7 of MRP-227, Revision 1-A, found acceptable as documented in the staff's safety evaluation dated April 25, 2019, provides the basis for implementing the program in accordance with NEI 03-08. Administrative activities for keeping the program implementation procedures up to date with the various industry reports within the scope of the AMP (e.g.,MRP-227, Revision 1-A) fall within the scope of this "Administrative Controls" program element. ~~The evaluation in Section 3.5 of the NRC's SE, Revision 1, on MRP-227-A provides the basis for endorsing NEI 03-08. This includes endorsement of the criteria in NEI 03-08 for notifying the NRC of any deviation from the I&E methodology in MRP-227-A and justifying the deviation no later than 45 days after its approval by a licensee executive.~~

10. ***Operating Experience****:* The review and assessment of relevant operating experience (OE) for its impacts on the program, including implementing procedures, are governed by NEI 03-08 and Appendix A of MRP-227, Revision 1-A. Consistent with MRP-227,

Revision 1-A, the reporting of inspection results and OE is treated as a "Needed" category item under the implementation of NEI 03-08.

The program is informed and enhanced when necessary through the systematic and ongoing review of both plant-specific and industry OE including research and development such that the effectiveness of the AMP is evaluated consistent with the discussion in Appendix B of the GALL-SLR Report.

## References

10 CFR Part 50, Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants."  Washington, DC: U.S. Nuclear Regulatory Commission.  2016.

10 CFR Part 50.55a, "Codes and Standards."  Washington, DC:  U.S. Nuclear Regulatory Commission.  2016.

ASME.  ASME Code, Section V, "Nondestructive Examination."  2004 Edition[2].  New York, New York:  American Society of Mechanical Engineers.

_____.  ASME Code, Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components."  New York, New York:  American Society of Mechanical Engineers.  2008.

EPRI.  EPRI Topical Report No. 1016596, "Materials Reliability Program: Pressurized Water Reactor Internals Inspection and Evaluation Guidelines (MRP-227, Revision 0)."  Palo Alto, California: Electric Power Research Institute.  2008.

_____.  EPRI ~~Technical~~ Topical Report No. 1022863, "Materials Reliability Program: Pressurized Water Reactor Internals Inspection and Evaluation Guidelines (MRP-227-A)." Agencywide Documents Access and Management System (ADAMS) Accession No. ML12017A193 (Transmittal letter from the EPRI-MRP) and ADAMS Accession Nos. ML12017A194, ML12017A196, ML12017A197, ML12017A191, ML12017A192, ML12017A195 and ML12017A199, (Final Report).  Palo Alto, California: Electric Power Research Institute.  December 2011.

_____.  EPRI 1016609, "Materials Reliability Program: Inspection Standard for PWR Internals (MRP-228)."  (Non-publicly available ADAMS Accession No. ML092120574).  The non-proprietary version of the report may be accessed by members of the public at ADAMS Accession No. ML092750569.  Palo Alto, California: Electric Power Research Institute. July 2009.

_____.  EPRI Topical Report 3002017168, "Materials Reliability Program: Pressurized Water Reactor Internals Inspection and Evaluation Guidelines (MRP-227, Revision 1-A)."  ADAMS Accession No. ML19339G350.  Palo Alto, California: Electric Power Research Institute. December 2019.

[2] GALL-SLR Report Chapter I, Table 1, identifies the ASME Code Section XI editions and addenda that are acceptable to use for this AMP.

NEI. NEI 03-08, Revision ~~2~~3, “Guideline for the Management of Materials Issues.” ADAMS Accession No. ML19079A253~~ML101050337~~. Washington, DC: Nuclear Energy Institute. ~~January 2010~~February 2017.

NRC. License Renewal Interim Staff Guidance LR-ISG-2011-04, “Updated Aging Management Criteria for Reactor Vessel Internal Components for Pressurized Water Reactors.” ADAMS Accession No. ML12270A436. Washington, DC: U.S. Nuclear Regulatory Commission. June 3, 2013.

_____. License Renewal Interim Staff Guidance LR-ISG-2011-05, “Ongoing Review of Operating Experience.” ADAMS Accession No. ML12044A215. Washington, DC: U.S. Nuclear Regulatory Commission. March 16, 2012.

_____. Safety Evaluation from Robert A. Nelson (NRC) to Neil Wilmshurst (EPRI), “Revision 1 to the Final Safety Evaluation of Electric Power Research Institute (EPRI) Report, Materials Reliability Program (MRP) Report 1016596 (MRP-227), Revision 0, Pressurized Water Reactor Internals Inspection and Evaluation Guidelines.” ADAMS Accession No. ML11308A770. Washington, DC: U.S. Nuclear Regulatory Commission. December 16, 2011.

_____. “Final Safety Evaluation for Electric Power Research Institute Topical Report MRP-227, Revision 1, ‘Materials Reliability Program: Pressurized Water Reactor Internals Inspection and Evaluation Guideline.’” ADAMS Accession No. ML19081A001. Washington, D.C: U.S. Nuclear Regulatory Commission. April 25, 2019.

_____. Letter from Joe Holonich (NRC) to Brian Burgos (EPRI), “U.S. Nuclear Regulatory Commission Verification Letter for Electric Power Research Institute Topical Report MRP-227, Revision 1, ‘Materials Reliability Program: Pressurized Water Reactor Internals Inspection and Evaluations Guideline.’” ADAMS Accession No. ML20006D152. Washington, D.C: U.S. Nuclear Regulatory Commission. February 19, 2020.

_____. Email from Joe Holonich (NRC) to Kyle Amberge (EPRI), “Transmittal Email MRP 227, Rev 1-A Supplemental Information -A Verification.” ADAMS Accession No. ML20175A149. Washington, D.C: U.S. Nuclear Regulatory Commission. July 7, 2020.

GALL-SLR Report Table XI-01 Proposed Revisions

**Table XI-01. FSAR Supplement Summaries for GALL-SLR Report Chapter XI Aging Management Programs**

| AMP | GALL-SLR Program | Description of Program | Implementation Schedule |
|---|---|---|---|
| XI.M16A | PWR Vessel Internals | The program relies on implementation of the inspection and evaluation guidelines in EPRI Technical Report No. ~~1022863~~ 3002017168 (MRP-227, Revision 1-A) and EPRI Technical Report No. 1016609 (MRP-228) to manage the aging effects on the reactor vessel internal components, as supplemented by a gap analysis that identifies enhancements to the program that are needed to address an 80-year operating period.<br><br>Alternatively, the program relies on implementation of an acceptable generic report such as an approved revision of MRP-227 that considers an operating period of 80 years<br><br>This program is used to manage (a) cracking, including stress corrosion cracking, primary water stress corrosion cracking, irradiation-assisted stress corrosion cracking, and cracking due to fatigue/cyclical loading; (b) loss of material induced by wear; (c) loss of fracture toughness due to either thermal aging, neutron irradiation embrittlement, or void swelling; (d) dimensional changes due to void swelling or distortion; and (e) loss of preload due to thermal and irradiation enhanced stress relaxation or creep.<br><br>[The applicant is to provide additional details to describe the gap analysis associated with the AMP.] | Program, accounting for the impacts of a gap analysis, is implemented 6 months prior to the subsequent period of extended operation, or alternatively, a plant-specific program may be implemented 6 months prior to the subsequent period to extended operation. |

## APPENDIX E

## PROPOSED REVISION TO GALL-SLR REPORT TABLE IX.C, "USE OF TERMS FOR MATERIALS"

NUREG-2191, Volumes 1 and 2, "Generic Aging Lessons Learned for Subsequent License Renewal (GALL-SLR) Report," Table IX.C, "Use of Terms for Materials," defines many generalized materials used in the aging management review tables in Chapters II through VIII of the GALL-SLR Report. The table below adds the term "stellite" and its usage to Table IX.C.

GALL-SLR Report Table IX.C Revisions

| **IX.C Use of Terms for Materials** | |
|---|---|
| **Term** | **Usage in this document** |
| Stellite | ASTM International provides a technical definition of stellite in ASTM MNL46, "Metallographic and Materialographic Specimen Preparation, Light Microscopy, Image Analysis and Hardness Testing":<br><br>"Stellite is a special cobalt-based alloy with 46–65 % Co, 25–25 % Cr, and 5–20 % W. The material is very wear resistant…" |

## APPENDIX F

## PROPOSED REVISIONS TO SRP-SLR TABLE 4.7-1, "EXAMPLES OF POTENTIAL PLANT-SPECIFIC TLAA TOPICS"

NUREG-2192, "Standard Review Plan for Review of Subsequent License Renewal Applications for Nuclear Power Plants" (SRP-SLR), Table 4.7-1, "Examples of Potential Plant-Specific TLAA Topics," provides examples of potential plant-specific time-limited aging analyses (TLAAs) that license renewal applicants have identified. This table is reproduced below in its entirety, with changes provided in redline format. This table supersedes SRP-SLR, Revision 0, Table 4.7-1.

SRP-SLR Table 4.7-1 Proposed Revisions

| **Table 4.7-1 Examples of Potential Plant-Specific TLAA Topics** |
|---|
| **BWRs** |
| Re-flood thermal shock of the reactor pressure vessel |
| Re-flood thermal shock of the core shroud and other reactor vessel internals |
| Loss of preload for core plate rim hold-down bolts |
| Erosion of the main steam line flow restrictors |
| Susceptibility to irradiation-assisted stress corrosion cracking |
| **PWRs** |
| Reactor pressure vessel underclad cracking |
| Leak-before-break |
| Reactor coolant pump flywheel fatigue crack growth |
| Response to NRC Bulletin 88-11, "Pressurizer Surge Line Thermal Stratification" |
| Response to NRC Bulletin 88-08, "Thermal Stresses in Piping Connected to Reactor Cooling Systems" |
| EPRI MRP cycle-based and fluence-based analyses in support of MRP-227 |
| **BWRs and PWRs** |
| Fatigue of cranes (crane cycle limits) |
| Fatigue of the spent fuel pool liner |
| Corrosion allowance calculations |
| Flaw growth due to stress corrosion cracking |
| Predicted lower limit |

## APPENDIX G

## LIST OF ABBREVIATIONS USED IN SLR-ISG-PWRVI-2020-XX

| | |
|---|---|
| ADAMS | Agencywide Document Access Management System |
| A/LAI | applicant/licensee action item |
| AMR | aging management review |
| AMP | aging management program |
| ANO-1 | Arkansas Nuclear One |
| ASME | American Society of Mechanical Engineers |
| BMI | bottom-mounted instrumentation |
| B&W | Babcock & Wilcox Company (currently part of the AREVA corporate complex of private companies) |
| CASS | cast austenitic stainless steel |
| CE | Combustion Engineering Company (currently owned by Westinghouse Electric Company) |
| CEA | control element assembly |
| CFR | *Code of Federal Regulations* |
| CRGT | control rod guide tube |
| CSS | core support shield |
| GALL | NUREG-1801, “Generic Aging Lessons Learned (GALL) Report” |
| GALL-SLR | NUREG-2191, Volumes 1 and 2, “Generic Aging Lessons Learned for Subsequent License Renewal Applications (GALL-SLR) Report” |
| FD | flow distributor |
| FSAR | final safety analysis report |
| EPRI | Electric Power Research Institute |
| IASCC | irradiation-assisted stress corrosion cracking |
| I&E | inspection and evaluation |

| | |
|---|---|
| IMI | incore monitoring instrument or incore monitoring instrumentation |
| ISG | interim staff guidance |
| LCB | lower core barrel |
| LR | license renewal |
| LRA | license renewal application |
| LTS | lower thermal shield |
| MRP | Materials Reliability Program |
| NRC | U.S. Nuclear Regulatory Commission |
| OE | operating experience |
| PH | precipitation hardened |
| PWR | pressurized-water reactor |
| PWSCC | primary water stress corrosion cracking |
| RIS | regulatory information summary |
| RVI | reactor vessel internal |
| SCC | stress corrosion cracking |
| SSC | structure, system, and component |
| SLR | subsequent license renewal |
| SLRA | subsequent license renewal application |
| SRP-LR | NUREG 1800, “Standard Review Plan for Review of License Renewal Applications for Nuclear Power Plants” |
| SRP-SLR | NUREG-2192, “Standard Review Plan for Review of Subsequent License Renewal Applications for Nuclear Power Plants” |
| SS | stainless steel |
| SSHT | surveillance specimen holder tube |
| TLAA | time-limited aging analysis |
| TR | topical report |

| | |
|---|---|
| UAW | upper axial weld (upper vertical weld) |
| UCB | upper core barrel |
| UTS | upper thermal shield |
| XL | extra-long |
| X-750 | generic reference to a type of nickel-based alloy metal that may be trademarked by industry manufacturers of the material |