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SLR-ISG-2021-01-PWRVI

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

Interim Staff Guidance

January 2021

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INTERIM STAFF GUIDANCE

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

SLR-ISG-2021-01-PWRVI

PURPOSE

The U.S. Nuclear Regulatory Commission (NRC) staff is issuing this 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 SLR-ISG identifies 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 June 2020 (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 SLR-ISG to accomplish the following five objectives:

(1) GALL-SLR Report and SRP-SLR Guidance Changes: 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) Clarification on the Use of MRP-227, Revision 1-A: 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) Reduction of Unnecessary Burden for PWR SLRAs: 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) Resolution of Applicant/Licensee Action Items (A/LAIs): 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) Closure of Regulatory Information Summary (RIS) 2011-07: 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 provides guidance to licensees that wish to extend their plant operating licenses from 60 years to 80 years, and SRP-SLR provides guidance 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 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 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 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 Clarification 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 to address anticipated aging effects associated with the requested 80-year operating period.

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 is not 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-2021-01-PWRVI.

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

Issuance of this ISG does not constitute a backfit as defined in 10 CFR 50.109(a)(1) and is not otherwise inconsistent with the issue finality provisions in 10 CFR Part 52, "Licenses, certifications, and approvals for nuclear power plants." Thus, the NRC staff did not prepare a backfit analysis for the issuance of this ISG.

The NRC staff's position is based upon the following considerations:

- The ISG positions do not constitute backfitting, inasmuch as the ISG is guidance directed to the NRC staff with respect to its regulatory responsibilities. The ISG provides interim guidance to the staff on how to review certain requests. Changes in guidance intended for use by only the staff are not matters that constitute backfitting as that term is defined in 10 CFR 50.109, "Backfitting," or that involve the issue finality provisions of 10 CFR Part 52.

- Backfitting and issue finality—with certain exceptions discussed in this section—do not apply to current or future applicants. Applicants and potential applicants are not, with certain exceptions, the subject of either the Backfit Rule or any issue finality provisions under 10 CFR Part 52. This is because neither the Backfit Rule nor the issue finality provisions of 10 CFR Part 52 were intended to apply to every NRC action that substantially changes the expectations of current and future applicants. The exceptions to the general principle are applicable whenever a 10 CFR Part 50 operating license applicant references a construction permit or a 10 CFR Part 52 combined license applicant references a license (e.g., an early site permit) or an NRC regulatory approval (e.g., a design certification rule) (or both) for which specified issue finality provisions apply. The NRC staff does not currently intend to impose the positions represented in this ISG in a manner that constitutes backfitting or is inconsistent with any issue finality provision of 10 CFR Part 52. If in the future the NRC staff seeks to impose positions stated in this ISG in a manner that would constitute backfitting or be inconsistent with these issue finality provisions, the NRC staff must make the requisite showing as set forth in the Backfit Rule or address the regulatory criteria set forth in the applicable issue finality provision, as applicable, that would allow the staff to impose the position.

- The NRC staff has no intention to impose the ISG positions on existing nuclear power plant licensees either now or in the future (absent a voluntary request for a change from the licensee). The staff does not intend to impose or apply the positions described in the ISG to existing (i.e., already issued) licenses (e.g., operating licenses and combined licenses). Hence, the issuance of this ISG—even if considered guidance subject to the Backfit Rule or the issue finality provisions in 10 CFR Part 52— would not need to be evaluated as if it were a backfit or as being inconsistent with issue finality provisions. If, in the future, the NRC staff seeks to impose a position in the ISG on holders of already issued licenses in a manner that would constitute backfitting or does not provide issue finality as described in the applicable issue finality provision, then the staff must make a showing as set forth in the Backfit Rule or address the criteria set forth in the applicable issue finality provision that would allow the staff to impose the position.

CONGRESSIONAL REVIEW ACT

This ISG is a rule as defined in the Congressional Review Act (5 U.S.C. 801-808). However, the Office of Management and Budget has not found it to be a major rule as defined in the Congressional Review Act.

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. Revisions to SRP-SLR Table 3.1-1

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

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

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

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

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

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

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

G. List of Abbreviations Commonly Used in SLR-ISG-2021-01-PWRVI

H. Disposition of Public Comments

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 Topical Report No. 3002017168, “Materials Reliability Program: Pressurized Water Reactor Internals Inspection and Evaluation Guidelines (MRP-227, Revision 1-A),” June 2020 (ADAMS Accession No. ML20175A112).

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 Topical 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 Topical 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

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"

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~~ ~~IV.B3.RP-357~~ ~~IV.B3.RP-400~~ IV.B2.RP-355 (if AMP XI.M16A is credited for aging management) 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) 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~~ ~~IV.B3.RP-382~~ ~~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 IV.B4.RP-241a IV.B4.RP-242a IV.B4.RP-247 IV.B4.RP-247a IV.B4.RP-248 IV.B4.RP-248a ~~IV.B4.RP-249a~~ ~~IV.B4.RP-252a~~ IV.B4.RP-252c IV.B4.RP-256 IV.B4.RP-256a ~~IV.B4.RP-258a~~ ~~IV.B4.RP-259a~~ IV.B4.RP-261 ~~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, 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 ~~IV.B4.RP-244a~~ IV.B4.RP-245 IV.B4.RP-245a IV.B4.RP-246 IV.B4.RP-246a IV.B4.RP-246c IV.B4.RP-246d ~~IV.B4.RP-250a~~ ~~IV.B4.RP-254~~ ~~IV.B4.RP-254a~~ IV.B4.RP-260a IV.B4.RP-262 IV.B4.RP-352
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 IV.B3.RP-314 IV.B3.RP-322 IV.B3.RP-324 ~~IV.B3.RP-326a~~ IV.B3.RP-327 IV.B3.RP-328 IV.B3.RP-338 IV.B3.RP-342 IV.B3.RP-343 IV.B3.RP-358 IV.B3.RP-362a ~~IV.B3.RP-363~~

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 IV.B3.RP-316 IV.B3.RP-323 IV.B3.RP-325 IV.B3.RP-329 IV.B3.RP-330 IV.B3.RP-333 IV.B3.RP-335 IV.B3.RP-362c 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 IV.B3.RP-320a 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 IV.B2.RP-271 IV.B2.RP-275 IV.B2.RP-276 ~~IV.B2.RP-280~~ IV.B2.RP-296a IV.B2.RP-298 IV.B2.RP-302 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 ~~IV.B2.RP-278~~ IV.B2.RP-280 IV.B2.RP-286 IV.B2.RP-291 IV.B2.RP-291a IV.B2.RP-291b IV.B2.RP-293 IV.B2.RP-294 IV.B2.RP-298a 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, 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 IV.B2.RP-301 IV.B2.RP-345a IV.B2.RP-346 IV.B2.RP-399 ~~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 IV.B3.RP-318 IV.B2.RP-326 IV.B3.RP-338a IV.B3.RP-359 IV.B3.RP-360 IV.B3.RP-362 ~~IV.B3.RP-364~~ IV.B3.RP-365 IV.B3.RP-366

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 IV.B3.RP-331 IV.B3.RP-333a IV.B3.RP-359a IV.B3.RP-361 IV.B3.RP-362b IV.B3.RP-364 IV.B3.R-455
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 IV.B3.RP-332 ~~IV.B3.RP-334a~~ IV.B3.RP-336 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 IV.B4.RP-240a IV.B4.RP-242 IV.B4.RP-247b IV.B4.RP-247c IV.B4.RP-248b IV.B4.RP-249 IV.B4.RP-251 IV.B4.RP-251a IV.B4.RP-252 IV.B4.RP-252b IV.B4.RP-256b IV.B4.RP-258 IV.B4.RP-259 ~~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-243 IV.B4.RP-243a IV.B4.RP-245b IV.B4.RP-245c IV.B4.RP-246b IV.B4.RP-246e IV.B4.RP-250 IV.B4.RP-252a ~~IV.B4.RP-254b~~ IV.B4.RP-260 IV.B4.RP-386

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 ~~IV.B2.RP-278a~~ IV.B4.RP-280a IV.B2.RP-287 IV.B2.RP-290 IV.B2.RP-290a IV.B2.RP-290b IV.B2.RP-292 IV.B2.RP-295 IV.B2.RP-297a 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 IV.B2.RP-288 IV.B2.RP-299 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 IV.B1.R-100 IV.B1.R-105 IV.B1.R-92 IV.B1.R-93 IV.B1.R-96 IV.B1.R-97 IV.B1.R-98 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 or LRA/SLRA-specified reactor vessel internal component 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 an adjusted site-specific or component-specific aging management basis for a specified reactor vessel internal component	Yes (SRP-SLR Section 3.1.2.2.9)	IV.B2.R-423 IV.B3.R-423 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, stellite PWR reactor vessel internal components or LRA/SLRA-specified reactor vessel internal component 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 an adjusted site-specific or component-specific aging management basis for a specified reactor vessel internal component	Yes (SRP-SLR Section 3.1.2.2.9)	IV.B2.R-424 IV.B3.R-424 IV.B4.R-424

APPENDIX B

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

APPENDIX B.1

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.

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 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	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.B2.RP-271	3.1-1, 053a	Baffle-to-former assembly: ~~accessible~~ baffle-to-former bolts (includes 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-272	3.1-1, 059a	Baffle-to-former assembly: ~~accessible~~ baffle-to-former bolts (includes 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

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-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
M	IV.B2.RP-275	3.1-1, 053a	Baffle-to-former assembly: baffle-edge bolts ~~(all plants with baffle-edge 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	REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM							
Table B2	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-354	3.1-1, 059a	Baffle-to-former assembly: baffle-edge bolts ~~(all plants with baffle-edge 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
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

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-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 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; loss of material due to wear	AMP XI.M16A, "PWR Vessel Internals"	Yes
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 toughness due to thermal aging embrittlement (CASS only)	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

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-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
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

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-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 plant-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~~
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

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~~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~~
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; changes in dimension due to void swelling or distortion	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-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
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

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-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
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)	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

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-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; changes in dimension due to void swelling or distortion	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
M	IV.B2.RP-290a	3.1-1, 059b	Lower ~~support~~internals assembly: lower support ~~forging or~~ casting	~~Stainless~~ Cast austenitic 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

IV REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM
Table B2 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-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
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

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-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
N	IV.B2.R-423	3.1-1, 118	Reactor vessel internal components or LRA/SLRA-specified reactor vessel internal 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), with an adjusted site-specific or component-specific aging management basis for a specified reactor vessel internal component	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**
N	IV.B2.R-424	3.1-1, 119	Reactor vessel internal components or LRA/SLRA-specified reactor vessel internal component	Stainless steel, nickel alloy, stellite (as a wear-resistant surface)	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 an adjusted site-specific or component-specific aging management basis for a specified reactor vessel internal component	Yes
~~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~~

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-302	3.1-1, 053a	Thermal shield assembly: thermal shield flexures	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.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

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.

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 Revisions

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-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 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-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~~D	~~IV.B3.RP-326a~~	~~3.1-1, 052a~~	~~Core shroud assembly (designs assembled in two vertical sections): assembly components, including monitoring of the gap opening at the core shroud re-entrant corners~~	~~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-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
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-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

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-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
M	IV.B3.RP-360	3.1-1, 056a	Core shroud assembly (for 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 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

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-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
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

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-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
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~~

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-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
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

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-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
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

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~~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~~
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

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-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
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 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**
N	IV.B3.R-423	3.1-1, 118	Reactor vessel internal components or LRA/SLRA-specified reactor vessel internal 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), with an adjusted site-specific or component-specific aging management basis for a specified reactor vessel internal component	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**
N	IV.B3.R-424	3.1-1, 119	Reactor vessel internal components or LRA/SLRA-specified reactor vessel internal component	Stainless steel, nickel alloy, stellite (as a wear-resistant surface)	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 an adjusted site-specific or component-specific aging management basis for a specified reactor vessel internal component	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 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**
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

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.

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 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 or fatigue	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, fatigue~~	~~AMP XI.M16A, "PWR Vessel Internals," and AMP XI.M2, "Water Chemistry"~~	~~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**
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 or fatigue~~	~~AMP XI.M16A, "PWR Vessel Internals," and AMP XI.M2, "Water Chemistry" (irradiation-assisted SCC only)~~	~~Yes~~
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

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-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
~~M~~D	~~IV.B4.RP-244a~~	~~3.1-1, 051b~~	~~Core barrel assembly: locking devices (including 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 SCCor fatigue~~	~~AMP XI.M16A, "PWR Vessel Internals," and AMP XI.M2, "Water Chemistry" (irradiation-assisted SCC only)~~	~~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**
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~~ or precipitation hardened (PH) stainless steel~~s~~	Reactor coolant and neutron flux	Loss of fracture toughness due to thermal aging embrittlement	AMP XI.M16A, "PWR Vessel Internals"	Yes
~~M~~D	~~IV.B4.RP-252a~~	~~3.1-1, 051a~~	~~Core support shield (CSS) 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~~

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-252a	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," and AMP XI.M2, "Water Chemistry" (SCC only)	Yes
N	IV.B4.RP-252b	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); 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-252c	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 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~~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~~D	~~IV.B4.RP-258a~~	~~3.1-1, 051a~~	~~Incore Monitoring Instrument (IMI) guide tube assembly: IMI guide tube spiders~~	~~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" (SCC and irradiation-assisted SCC 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-258	3.1-1, 058a	Incore Monitoring Instrument (IMI) guide tube assembly: IMI ~~Incore~~ guide tube spiders ~~(castings)~~	Stainless steel, including CASS	Reactor coolant and neutron flux	Loss of fracture toughness due to ~~thermal aging and~~ neutron irradiation embrittlement or thermal aging embrittlement (for spiders made from CASS)	AMP XI.M16A, "PWR Vessel Internals"	Yes
~~M~~D	IV.B4.RP-259a	~~3.1-1, 051a~~	~~Incore Monitoring Instrument (IMI) guide tube assembly: IMI guide tube spider-to-lower grid rib sections welds~~	~~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" (SCC and irradiation-assisted SCC 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 section~~s~~ 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, including alternate weld configuration at 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 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-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~~
~~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/stud 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/stud 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

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-246	3.1-1, 051b	Lower grid assembly: ~~upper thermal shield (UTS) bolts and~~ lower thermal shield (LTS) bolts or studs/nuts	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
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)~~ 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	Loss of fracture toughness due to neutron irradiation 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**
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 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-423	3.1-1, 118	Reactor vessel internal components or LRA/SLRA-specified reactor vessel internal 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), with an adjusted site-specific or component-specific aging management basis for a specified reactor vessel internal component	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 LRA/SLRA-specified reactor vessel internal component	Stainless steel, nickel alloy, stellite (as a wear-resistant surface material)	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 an adjusted site-specific or component-specific aging management basis for a specified reactor vessel internal component	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~~ including alternate weld configuration at 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 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-386	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

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.

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 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

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 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. ML20175A112). 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~~ and (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~~. ML20175A112~~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), but identified a new A/LAI.

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

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 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~~due to 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 MRP-227, Revision 1-A, are identified and justified for each component in the applicable

[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.

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~~due to SCC, PWSCC, IASCC, or fatigue/cyclic loading; (b) loss of material ~~induced by~~due to wear; (c) loss of fracture toughness ~~induced by~~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.

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, Revision 1-A, or 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-A, or as modified by a gap analysis~~MRP-227 (as supplemented).

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, Revision 1-A ~~-A (as supplemented)~~ and its subsections, or MRP-227 (as supplemented). Component reinspection frequencies for “Primary” and “Expansion” category components are defined in specific tables in Section 4 of the MRP-227, Revision 1-A report or in MRP-227 (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, Rev. 3 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, Revision 1-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, or MRP-227 (as supplemented) 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, Revision 1-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.

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

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)." ADAMS Accession No. ML090160206. Palo Alto, California: Electric Power Research Institute. December 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 Proprietary Topical Report No. ~~1016609~~3002010399, "Materials Reliability Program: Inspection Standard for PWR Internals (MRP-228, Rev. 3)." (Non-publicly available ADAMS Accession No. ~~ML092120574~~ML19081A064). The non-proprietary version of the report may be accessed by members of the public at ADAMS Accession No. ~~ML092750569~~ML19081A058. Palo Alto, California: Electric Power Research Institute. ~~July 2009~~November 2018.

_____. EPRI Topical Report 3002017168, "Materials Reliability Program: Pressurized Water Reactor Internals Inspection and Evaluation Guidelines (MRP-227, Revision 1-A)." ADAMS

[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.

Accession No. ML20175A112. Palo Alto, California: Electric Power Research Institute. June 2020.

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.

GALL-SLR Report Table XI-01 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. 3002010399~~1016609~~ (MRP-228, Rev. 3) 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. 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 This program is used to manage (a) cracking~~, including~~ due to 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~~due to 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. [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

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”: “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

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 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-2021-01-PWRVI

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

APPENDIX H

Disposition of Public Comments

Comments received regarding the draft version of this interim staff guidance are available electronically at the U.S. Nuclear Regulatory Commission's (NRC's) electronic Reading Room at http://www.nrc.gov/reading-rm/adams.html. From this page, the public can access the Agencywide Documents Access and Management System (ADAMS), which provides text and image files of publicly available documents. The following table lists the comments the NRC received regarding the draft version of this ISG.

Letter Number	ADAMS Accession No	Commenter Affiliation	Commenter Name
1	ML20246G654	Nuclear Energy Institute (NEI)	Peter W. Kissinger
2	ML20245E539	Electric Power Research Institute, Materials Reliability Program (EPRI MRP)	Christopher Koehler and Brian Burgos

As indicated in the table above, the staff received two public comment letters. The comment set submitted by NEI includes a total of three comments on the contents of the draft ISG. The comment set from the EPRI MRP submitted comments of behalf of Mr. J. McKinley (Westinghouse Electric Company, providing a total of 27 comments), Mr. E. Blocher (Dominion Energy Company; submitting one comment), and Mr. M. DeVan (Framatome Corporation, submitting five comments).

The table that follows on the next page of this Appendix provides the comment sources and numbers as listed in the public comment letter, the original comment(s) as written by the commenter, and the NRC staff's response to a specified comment or to a group of comments that the staff compiled as being similar in context. Some comments provided by NEI or the EPRI MRP include justifications for the comments or proposed actions for consideration by the staff for resolution of the specific comments. These justifications and proposed actions may be reviewed through access to ADAMS in the NRC's public electronic Reading Room (https://www.nrc.gov/reading-rm/adams.html) and performing a Web-based ADAMS search (WBA search) for the ML numbers associated with the comment source documents listed in the table above.

Disposition of Public Comments

Comment #(s)	ISG Section/Page	Comment	NRC Staff Response
Comments From NEI			
NEI #1	ISG Appendix C, Edits to AMR Further Evaluation Section 3.1.3.2.9 (ISG pages 3 and 4 of 4 in the appendix)	NEI wrote: "PWR applicants will use MRP-191, Rev 2, "Screening, Categorization, and Ranking of Reactor Internals Components, for Westinghouse and Combustion Engineering PWR Design," or MRP-189 Rev 3, "Screening, Categorization, and Ranking of Babcock and Wilcox Designed Pressurized Water Reactor Internals Component Items and Welds," as the principal input for the aging effects and aging mechanism screening portion (first two bullets of the Gap Analysis on page 3 of Appendix C) for their Reactor Vessel Internals (RVI) Gap Analysis. Recommend revising the ISG to reference MRP-191 Rev 2 or MRP-189 Rev 3 as one acceptable way to screen RVI aging effects and aging mechanisms (first two bullets of the Gap Analysis on page 3 of ISG Appendix C)."	The staff did not accept NEI Comment #1 or NEI's recommended revision of the staff's AMR Further Evaluation guidance in SRP-SLR Sections 3.1.2.2.9 and 3.1.3.2.9 (i.e., to include guidance criteria that would permit use of the proprietary EPRI MRP-191, Rev. 2 or MRP-189, Rev. 3 reports for component-specific screening objectives). Specifically, these reports have not been formally reviewed for acceptance by the NRC staff and were not used as the component-specific screening report criteria for the staff-accepted inspection and evaluation (I&E) guidelines in MRP-227, Rev. 1-A. Instead, these reports form the screening and ranking bases for what will be EPRI's updated I&E guidelines for a planned MRP-227, Rev. 2 report, which has yet to be submitted for staff review. Accordingly, the staff does not find it appropriate to reference MRP-191, Rev. 2 or MRP-189, Rev. 3 in the staff's updates of SRP-SLR Sections 3.1.2.2.9 or 3.1.3.2.9, as provided in Appendix C of this ISG. However, because the staff's update of AMP XI.M16A, "PWR Vessel Internals," in the ISG permits the use of additional reports or methodologies to supplement MRP-227, Rev. 1-A, an SLR applicant would not be precluded from using MRP-191, Rev. 2 or MRP-189, Rev. 3 for component-specific screening objectives if the applicant determines that the use of those reports is appropriate for its RVI management program. For SLR applicants that decide to use these reports, the staff expects that use of the reports would be discussed, evaluated, and supported in the applicant's technical basis document for its RVI management program.

Comment #(s)	ISG Section/Page	Comment	NRC Staff Response
NEI #2	ISG Appendix C, Edits to AMR Further Evaluation Section 3.1.3.2.9 (ISG pages 3 and 4 of 4 in the appendix)	NEI wrote: "MRP 2018-022, "Transmittal of MRP-191-SLR Screening, Ranking and Categorization Results and Interim Guidance in Support of Subsequent License Renewal at U.S. PWR Plants," provides expert panel results that supplement MRP-227 Inspection and Evaluation guidance for subsequent license renewal prior to the publication of MRP-227 Revision 2. Recommend revising the ISG to reference MRP-2018-022 as an acceptable starting point to identify changes to the existing MRP-227 Rev 1-A program characteristics or criteria (third bullet of the Gap Analysis on page 3 of ISG Appendix C)."	The staff did not accept NEI Comment #2 or NEI's recommended revision of the staff's AMR Further Evaluation guidance in SRP-SLR Sections 3.1.2.2.9 and 3.1.3.2.9 (i.e., to state that EPRI Report MRP-2018-022 may be used to identify component-specific gap analysis changes to the I&E criteria defined for the components in the MRP-227, Rev. 1-A report). The staff acknowledges that the MRP-2018-022 report was developed by EPRI as an Expert Panel basis to identify those changes in I&E criteria in MRP-277, Rev. 1-A (as developed under a 60-year assessment basis) that would be necessary for a subsequent period of extended operation (i.e., by further licensed operations for years 60 – 80). The staff also acknowledges that MRP-2018-022 was used and acceptably justified on a case-by-case basis for the RVI gap analyses of the RVI management programs in the first two PWR SLRAs reviewed by the staff (i.e., those for the nuclear units at the Turkey Point and Surry power stations). However, MRP-2018-022 is currently limited in that it only applies to Westinghouse and CE design PWR vessel internals and the scope of the report does not include RVI components in B&W design PWRs. Nor has NRC staff review of MRP-2018-022 been requested. Thus, the staff has not found the report acceptable for generic use by the industry licensees. Based on this rationale, the staff does not find it appropriate to reference MRP-2018-022 in the staff's updates of SRP-SLR Sections 3.1.2.2.9 and 3.1.3.2.9, as provided in Appendix C of the ISG. However, since the staff's update of GALL-SLR AMP XI.M16A, "PWR Vessel Internals," in the ISG permits use of additional reports or methodologies in supplement of MRP-227, Rev. 1-A, this would not preclude an SLR applicant from using MRP-2018-022 for the gap analysis of its PWR internal components if it is determined that the use of the report is appropriate for the RVI management program. For SLR applicants that decide to apply this report as one of the supporting reports for the RVI management program, the staff expectation is that use of the report would be discussed, evaluated, and supported in the applicant's technical basis document for the program.

Comment #(s)	ISG Section/Page	Comment	NRC Staff Response
NEI #3	ISG Appendices B1 through B4, various pages	NEI wrote: "Aging effects and mechanisms identified in NUREG-2191 AMR lines and NUREG-2192 Table 1s should be consistent with those identified in MRP-227 Rev 1-A Section 4 I&E Tables."	The staff partially accepted NEI's comment. The staff acknowledges that, for the development of SRP-SLR and GALL-SLR Report, the norm was to include an AMR item for cracking of a specified component with an AMR item on irradiation or thermal embrittlement of the component even if cracking mechanisms were not attributed as being applicable to the component in Chapter 4 of MRP-227, Rev. 1-A. The reason for this is that the inspection methods implemented by the AMP and by MRP-227, Rev. 1-A cannot inspect for direct evidence of embrittlement, and instead are only designed to look for presence of flaw indications that, if detected, may provide indirect evidence of embrittlement occurring in the components. Besides those listed in Chapter 4 of MRP-227, Rev. 1-A, the staff's identified aging effects and mechanisms in the AMR items of SLR-ISG-2021-01-PWRVI are also based on lessons learned from the gap analyses provided in the Turkey Point and Surry SLRAs or as cited in the SLRAs as being contained in the MRP-2018-022, 80-Year Expert Panel report. Additionally, for some cases, the Chapter 4 AMR item in MRP-227, Rev. 1-A for a given component may cite cracking as an applicable aging effect without listing the cracking mechanisms. For these cases, the staff used its own engineering judgement for the cited cracking mechanisms. Based on partial acceptance of this comment, the staff has reviewed the relevant AMR items for the accuracy of cited aging effects and mechanisms and adjusted the cited effects and mechanisms in the applicable AMR items on a case-by-case basis. The staff will provide the final basis for the cited aging effects and mechanisms in the staff's official technical basis statements for the specified AMR items.

Comment #(s)	ISG Section/Page	Comment	NRC Staff Response
Comments from the EPRI MRP			
EPRI MRP #1 and #8	For #1, ISG Appendix A, Table 3.1-1, AMR Item 053a (Page 5 of the appendix) For #8, ISG Appendix B.1, GALL-SLR Table IV.B2, Item IV.B2.RP-296a (Page 6 of the appendix)	Collectively, EPRI MRP made the following statements, as a basis for deleting GALL-SLR IV.B2.RP-296a as a referenced GALL-SLR item for SRP-SLR Table 3.1-1, Item 053a in Appendix A of the ISG and for deleting GALL-SLR Item IV.B2.RP-296a from the scope of Table IV.B2 in Appendix B1 of the ISG: “IV.B2.RP-296a adds the guide plates/cards for cracking mechanisms. This has been dispositioned in the past (See MRP-227 Table 3-3) and the same is expected for the future. The mechanisms are not new for SLR, so this does not seem like a necessary addition.” “It is true that SCC and fatigue were screened in for the guide cards/plates in earlier revisions, as well, but Table 3-3 in MRP-227 shows that these mechanisms were dispositioned as no additional measures. One major reason for this was the inclusion of the lower flange welds as leading components that are directly representative of the guide plates/cards.”	The staff did not accept EPRI MRP Comments #1 and #8 or the EPRI MRP’s statements that: (1) cracking should not be attributed for guide plates (guide cards) in Westinghouse-design control rod guide tube (CRGT) assemblies, and (2) GALL-SLR Report item IV.B2.RP-296a should not be included as a new AMR item in Appendix B.1 of the ISG or referenced as a GALL-SLR Report item for the update of SRP-SLR Table 3.1-1, Item 053a in Appendix A of the ISG. Specifically, based on lessons learned from the staff’s approval of the first two (2) SLRAs for the Westinghouse-designed PWR units (i.e., the Turkey Point and Surry SLRAs), cracking was identified as potentially applicable to the AMR assessments of the CRGT guide cards in the units. This included reporting of stress corrosion cracking (SCC) as a potentially applicable cracking mechanism for the CRGT cards in the Turkey Point units and fatigue as an applicable cracking mechanism for the CRGT guide card in the Surry units. Thus, the staff does not find EPRI MRP Comments #1 and #8 to be consistent with industry lessons learned from the staff’s processing of the first two SLRAs for Westinghouse-design PWR units. GALL-SLR Report Item IV.B2.RP-296a will remain as a new GALL-SLR Report item for managing cracking in Westinghouse-design CRGT guide plates (guide cards), as cited in Appendix B.1 of the ISG, and will be referenced in the staff’s update of SRP-SLR Table 3.1-1, Item 053a in Appendix A of the ISG. However, there is no requirement for an applicant to use the referenced GALL-SLR Report AMR item IV.B2.RP-296a for its SLRA if the applicant’s integrated plant assessment (IPA) for the SLRA concludes that cracking is not an aging effect that requires management for the specified CRGT guide card components.

Comment #(s)	ISG Section/Page	Comment	NRC Staff Response
EPRI MRP #2 and #10	For #2, ISG Appendix A, Table 3.1-1, AMR Item 053c (Page 6 of the appendix) For #10, ISG Appendix B.1, GALL-SLR Table IV.B2, Item IV.B2.RP-345a (Page 8 of the appendix)	Collectively, EPRI MRP made the following statements as a basis for deleting GALL-SLR Report AMR item IV.B2.RP-345a as a referenced item in SRP-SLR Table 3.1-1, item 053c, and for deleting GALL-SLR Report item IV.B2.RP-345a from Table IV.B2 in Appendix B1 of the ISG: Comment #2: "IV.B2.RP-345a adds cracking mechanisms for the upper core barrel flange. This has been dispositioned in past MRP-227 revisions because the UFW is the location of concern for cracking (due to the weld) and is already included as a Primary." Comment #10: "Item IV.B2.RP-345a is covered by the cracking screened in for the UFW in IV.B2.RP-276 (see overall comment 2). Cracking is most likely to occur at the weld and not on the base metal of the flange."	The staff did not accept EPRI MRP Comments #2 and #10 or the EPRI MRP's statements that: (1) cracking should not be attributed for core barrel flanges (base metal of components) in Westinghouse-design core barrel assemblies and (2) GALL-SLR Report item IV.B2.RP-345a should not be included as a new AMR item in Appendix B.1 of the ISG or referenced as a GALL-SLR Report AMR item for the update of SRP-SLR Table 3.1-1, item 053c in Appendix A of the ISG. Specifically, based on lessons learned from the staff's processing and approval of the first two SLRAs for the Westinghouse-designed PWR units (i.e., the Turkey Point and Surry SLRAs), cracking was identified as potentially applicable to the AMR assessments of the core barrel flanges in the units, including the reporting of fatigue as an applicable cracking mechanism for the core barrel flanges of the Surry units. Thus, the staff does not find EPRI MPR Comments #2 and #10 to be consistent with industry lessons learned from the staff's processing of these SLRAs. GALL-SLR Item IV.B2.RP-345a will remain as a new GALL-SLR item for managing cracking in Westinghouse-design core barrel flanges, as cited in Appendix B.1 of the ISG and will be referenced in the staff's update of SRP-SLR Table 3.1-1, Item 053c in Appendix A of the ISG. However, there is no requirement for an applicant to use the referenced GALL-SLR Report AMR item IV.B2.RP-345a for its SLRA if the applicant's integrated plant assessment (IPA) for the SLRA concludes that cracking is not an aging effect that requires management for the specified core barrel flange components.

Comment #(s)	ISG Section/Page	Comment	NRC Staff Response
EPRI MRP #3	ISG Appendix B.1, GALL-SLR Table IV.B2, Items IV.B2.RP-299 and IV.B2.RP-301 (Page 2 of the appendix)	EPRI MRP wrote the following as a basis for requesting adjustments of the component descriptions in GALL-SLR Items IV.B2.RP-299 and IV.B2.RP-301: “The addition of "(fuel alignment pins)" to this line is not correct, since the UCP alignment pins are different than the fuel alignment pins. The UCP alignment pins are attached to the core barrel and interface with the alignment slots on the edge of the upper core plate. It is possible that the inclusion of reference to TB-16-4 here in the Existing table (Table 4-9) has caused some confusion. This was intended to incorporate the latest OE to MRP-227 by reference. However, that technical bulletin does not provide requirements for the UCP alignment pins themselves. (See SLR new existing component - wear - MRP 2018-022).”	The staff accepted EPRI MRP Comment #3. Upon further review of the past Turkey Point and Surry SLRAs, the staff confirmed the fuel alignment pins in the upper internals assembly are not the same components as the core plate alignment pins in Westinghouse-design PWRs. The staff has deleted the parenthetical clause “(fuel alignment pins)” from the “Component” column entries in the updated versions of GALL-SLR Items IV.B2.RP-299 and IV.B2.RP-301, as updated in Appendix B.1 of the ISG. In addition, “loss of fracture toughness due to neutron irradiation embrittlement” has been removed as an applicable aging effect and mechanism combination for the core plate alignment pins in GALL-SLR Report Item IV.B2.RP-299, as loss of material due to wear is listed as the only non-cracking aging effect and mechanism combination for the core plate alignment pins in Item W15 of Table 4.9 in the MRP-227, Rev. 1-A report. Thus, the modification of GALL-SLR Report item IV.B2.RP-299 has been deleted from the final ISG, and no change is proposed to IV.B2.RP-299.
EPRI MRP #4	ISG Appendix B.1, GALL-SLR Table IV.B2, Items IV.B2.RP-275 and IV.B2.RP-354 (Page 4 of the appendix)	EPRI MRP wrote the following as a basis for requesting adjustments of the component descriptions in GALL-SLR Items IV.B2.RP-275 and IV.B2.RP-354: “The modified text deleted "all plants with baffle-edge bolts" and replaced it with "corner bolts". This is not correct. Corner bolts are a subset of baffle-former bolts, not baffle-edge bolts. Note that Bracket bolts are a subset of baffle-edge bolts. (See MRP-227 Table 4-3 W7-Baffle Former Assembly (Includes: Baffle plates, baffle edge bolts, corner bolts).”	The staff accepted EPRI MRP Comment #4. Based on the staff’s acceptance of EPRI’s rationale made in Comment #4, the staff deleted reference of corner bolts and the component-related parenthetical explanation from the “Structure and/or Component” column entries in the final versions of the GALL-SLR IV.B2.RP-275 and IV.B2.RP-354 items in the ISG. The modified “Structure and/or Component” column entries for the “RP-275” and “RP-354” items now read as “*Baffle-to-former assembly: baffle-edge bolts*”. Based on the statements in EPRI MPR Comment #4 and the staff’s discussions of the comment with members of the industry during the public meeting on the ISG of November 19, 2020, the staff also modified the “Structure and/or Component” column entries of the GALL-SLR Report AMR items IV.B2.RP-271 and IV.B2.RP-272 for baffle-to-former bolts to include the parenthetical clause “*(includes corner bolts)*” as part of the component descriptions.

Comment #(s)	ISG Section/Page	Comment	NRC Staff Response
EPRI MRP #5 and #6	For #5, ISG Appendix B.1, GALL-SLR Table IV.B2, Item IV.B2.RP-293 For #6, ISG Appendix B.1, GALL-SLR Table IV.B2, Item IV.B2.RP-292 (Page 5 of the appendix)	EPRI MRP wrote the following as a basis for requesting deletion of the mechanism IASCC from GALL-SLR Report AMR item IV.B2.RP-293 and the aging effect and mechanism combination of changes in dimension due to void swelling or distortion from GALL-SLR Report AMR item IV.B2.RP-292: "Not sure where IASCC comes from for the BMI column bodies. These are only screened in for SCC, wear, and fatigue in MRP-191, Rev. 2. They are only included for fatigue and IE in MRP-227, Rev. 1-A." "Similar to Item IV.B2.RP-293, the reason for adding void swelling, distortion, and changes in dimension [i.e., to Item IV.B2.RP-292] is not clear. This does not show up in MRP-191 or MRP-227, Rev. 1-A."	The staff accepted EPRI MRP Comments #5 and #6 for the final adjustment adjustments of GALL-SLR Report items IV.B2.RP-292 and IV.B2.RP-293 in Appendix B.1 of the ISG. Specifically, the staff confirmed that irradiation-assisted stress corrosion cracking (IASCC) and void swelling (VS) are not listed as applicable irradiation-induced aging mechanisms for BMI column bodies in item W2.2 of Table 4-6 in the MRP-227, Rev. 1-A reports. Although the staff confirmed that the Surry SLRA had indicated that IASCC and VS were identified as applicable mechanisms for the BMI column bodies in MRP-191, Rev. 1, it also identified that the IASCC and VS mechanisms were moved from the assessment of the components based on the 80-Year Expert Panel basis for the column bodies performed in MRP-2018-022. Thus, the staff finds this basis to be sufficient for removing "IASCC" as a listed aging mechanism for the update of GALL-SLR Item IV.B2.RP-293 in Appendix B.1 of the ISG and for removing "changes in dimension due to void swelling or distortion" as a listed aging effect and mechanism combination for the components in GALL-SLR Item IV.B2.RP-292. The "Aging Effect/Mechanism" column entry for GALL-SLR Item IV.B2.RP-293 will now read as "*Cracking due to SCC or fatigue*" for the final version of the line item in Appendix B.1 of the ISG. Similarly, the "Aging Effect/Mechanism" column entry for GALL-SLR Report item IV.B2.RP-292 will now read as "*Loss of fracture toughness due to neutron irradiation embrittlement; loss of material due to wear*".

Comment #(s)	ISG Section/Page	Comment	NRC Staff Response
EPRI MRP #7	ISG Appendix B.1, GALL-SLR Table IV.B2, Item IV.B2.RP-296 (Page 6 of the appendix)	EPRI MRP wrote the following as a basis for requesting deletion of loss for fracture toughness due to irradiation embrittlement from GALL-SLR Item IV.B2.RP-296: “Not sure where the IE comes from here for the guide cards/plates. This mechanism does not show up in MRP-227, Rev. 1-A or MRP-191, Rev. 2. It was included for the CASS guide cards in MRP-191, Rev. 1, but the updated and refined analysis in MRP-191, Revision 2 did not show IE for these.”	The staff accepted EPRI MRP Comment #7 for the final adjustment of GALL-SLR Report item IV.B2.RP-296 in the ISG and the EPRI MRP’s statement in the comment that GALL-SLR Report item IV.B2.RP-296 should not list irradiation embrittlement (IE) as a listed loss of fracture toughness mechanism for the guide plates (guide cards) in Westinghouse-design control rod guide tube (CRGT) assemblies. Guide cards made from CASS will still be listed as being susceptible to loss of fracture toughness due to thermal aging embrittlement (TE). Based on acceptance of EPRI MRP Comment #7, the staff has amended “Aging Effect/Mechanism” column entry for GALL-SLR Report item IV.B2.RP-296 to read as “*Loss of material due to wear; loss of fracture toughness due to thermal aging embrittlement (CASS only)*”.
EPRI MRP #9	ISG Appendix B.1, GALL-SLR Table IV.B2, Item IV.B2.RP-355 (Page 7 of the appendix)	EPRI wrote the following in relation to the staff’s proposed revision of “Aging Management Program (AMP)/TLAA” column entry for GALL-SLR Item IV.B2.RP-355 in the ISG: “Should the statement "using component-specific evaluation per MRP guidelines" be "using plant-specific aging management program per MRP-227 guidelines" to be consistent with MRP-227, Revision 1-A, Section 4.5?”	The staff accepted EPRI MRP Comment #9 for the final adjustment of GALL-SLR Report item IV.B2.RP-355 in the ISG and the EPRI MRP’s recommended editorial change of GALL-SLR Report item IV.B2.RP-355 as stated in the comment. Section 4.5 in the MRP-227, Rev. 1-A report uses the words “plant-specific,” so the staff has amended the wording to be consistent with those stated in the MRP-227, Rev. 1-A report. The “Aging Management Program (AMP)/TLAA” column entry of GALL-SLR Report item IV.B2.RP-355 has been amended to read as “*AMP XI.M16A, "PWR Vessel Internals," and AMP XI.M2, "Water Chemistry" (for SCC mechanisms only) – using plant-specific evaluation per MRP guidelines*”.

Comment #(s)	ISG Section/Page	Comment	NRC Staff Response
EPRI MRP #11 and #12	ISG Appendix B.1, GALL-SLR Table IV.B2, Items IV.B2.RP-280 and IV.B2.RP-280a (Page 9 of the appendix)	EPRI wrote the following as a basis for requesting removal of irradiation stress corrosion cracking (IASCC) as a listed aging mechanism for core barrel lower flange welds (LFWs) in the updated ISG version of GALL-SLR Items IV.B2.RP-280 and for deleting GALL-SLR IV.B2.RP-280a from the scope of the ISG: "The grouping of the LGW and LFW in MRP-191, Revision 2 have likely caused confusion here. The LGW is in the core beltline and is subject to irradiation effects, such as IASCC. The LFW is located near the bottom of the core barrel where IASCC is not an effect. These were grouped originally because they are both in the lower core barrel. Note that this type of detail will be addressed in MRP-232 and MRP-227, Rev. 2." "Similar to overall comment 11 above, the LFW is far from the core and not susceptible to irradiation effects."	The staff did not accept EPRI MRP Comments #11 and #12 or the EPRI MRP's collective statements in the two comments that neither irradiation-assisted stress corrosion cracking (IASCC) nor irradiation embrittlement (IE) should be attributed to Westinghouse design core barrel lower flange welds (LFWs). Based on lessons learned from the staff's review of the Surry SLRA, the past applicant indicated that IASCC, IE, and void swelling (VS) were all attributed as being screened-in irradiation mechanisms for Westinghouse-design core barrel LFWs per EPRI's 80-Year Expert Panel analysis of the components in MRP-2018-022. Specifically, the Surry SLRA indicated that the core barrel LFWs are within 80-year fluence exposure zones high enough to screen the welds in for IASCC, IE and VS mechanisms. Thus, the final version of GALL-SLR Report item IV.B2.RP-280 has been amended to include IASCC as a potential aging mechanism for the core barrel LFWs. Similarly, the final version of GALL-SLR Report item IV.B2.RP-280a in the ISG has been amended further to include both IE and VS as potential aging mechanisms for the core barrel LFWs based on the inclusion of IE and VS in the Surry gap analysis. The staff cannot rely on MRP-227, Rev. 2 report as the basis for providing updated aging mechanisms for the LFWs, as the report has not yet to be docketed for staff approval in ADAMS or accepted by the staff for implementation. Given these comment considerations, the "Aging Effect/Mechanism" column entry for GALL-SLR Report item IV.B2.RP-280 will remain as amended in the ISG to state "*Cracking due to SCC, IASCC (lower flange weld only), or fatigue*," and the corresponding column entry in GALL-SLR Report item IV.B2.RP-280a has been further amended in the final ISG to state "*Loss of fracture toughness due to neutron irradiation embrittlement; changes in dimension due to void swelling or distortion*."

Comment #(s)	ISG Section/Page	Comment	NRC Staff Response
EPRI MRP #13	ISG Appendix B.1, GALL-SLR Table IV.B2, Items IV.B2.RP-287, IV.B2.RP-290, IV.B2.RP-295 (Pages 12 and 13 of the appendix)	EPRI wrote the following as a basis for deleting the additional aging mechanisms cited in the staff's proposed updated of GALL-SLR Item IV.B2.RP-287: "A problem in multiple places is the addition of all degradation mechanisms screened in under MRP-191, Rev. 2. For example, the LSC bolts and support columns were screened in for VS under MRP-191, Rev. 1; however, those were dispositioned for MRP-227 based on more detailed evaluation of those locations. Based on that, VS was assigned to "no additional measures" in Table 3-3 of MRP-227, Revision 1-A. That same logic applies here, too." The comment also applied to GALL-SLR Items IV.B2.RP-290 and IV.B2.RP-295, as updated in Appendix B.1 of the ISG.	The staff did not accept EPRI MRP Comment #13 or the EPRI MRP's statement in the comment that void swelling or distortion should not be included as cited aging mechanisms for the non-cracking based AMR line items for Westinghouse lower support column bodies made from either cast or wrought stainless steel materials (i.e., in GALL-SLR Report items IV.B2.RP-290 or IV.B2.RP-295) and for lower support column bolts made from stainless steel (i.e., in GALL-SLR Report item IV.B2.RP-287). Specifically, based on lessons learned from the staff's review of the Surry SLRA, the past applicant indicated that void swelling (VS) was attributed as being a screened-in an irradiation aging mechanism for Westinghouse-design lower support column bodies and lower support column bolts per EPRI's 80-Year Expert Panel analysis of the components in MRP-2018-022. Specifically, the Surry SLRA indicates that the lower support column bodies and lower support column bolts are within 80-year fluence exposure zones high enough to screen the components in for applicable irradiation-mechanisms (i.e., irradiation-assisted stress corrosion cracking (IASCC), irradiation embrittlement (IE), and VS, and for the bolts, irradiation-enhanced stress relaxation/irradiation-enhanced creep (ISR/IC) as a potential loss of preload mechanism). Based on these considerations, the staff finds it appropriate for the updates of the GALL-SLR Report items IV.B2.RP-290 and IV.B2.RP-295 for the cast and non-cast lower support column bodies to cite VS and IE as applicable non-cracking, irradiation-based mechanisms for the column bodies, and for the update of GALL-SLR Report item IV.B2.RP-287 for the lower support column bolts to include VS, IE, and ISR/IC as the listed non-cracking, irradiation-based mechanisms for the bolts.

EPRI MRP #14, #15, #16, #17, #22 and #23	For #14 and #16, ISG Appendix B.1, GALL-SLR Table IV.B2, Items IV.B2.R-423 and IV.B2.R-424 (Page 14 of ISG Appendix B.1) For #15 and #17, ISG Appendix A, SRP-SLR Table 3.1-1, Items 118 and 119 (Pages 10 and 11 of ISG Appendix A) For #22 and #23, ISG Appendix B.2, GALL-SLR Table IV.B3, Items IV.B3.R-423 and IV.B3.R-424 (Pages 14 and 15 of ISG Appendix B.2)	For the updates of GALL-SLR Items IV.B2.R-423 and IV.B2.R-424 in ISG Appendix B.1 and GALL-SLR Items IV.B3.R-423 and IV.B3.R-424 in ISG Appendix B.2, EPRI wrote: “This is a catch-all AMR line. Revise the AMP column to delete "or specified reactor internal component-specific aging management basis". The Structure or component column already indicates the need for a site-specific or component specific aging management basis.” For the updates of SRP-SLR Table 3.1-1 Items 118 and 119 in ISG Appendix A, EPRI wrote: “Revise Table 3.1-1 item 118 Component column and the Aging Management Program/TLAA column to be consistent with revision of AMR IV.B2.R-423.” “Revise Table 3.1-1 item 119 Component column and the Aging Management Program/TLAA column to be consistent with revision of AMR IV.B2.R-424.”	The staff partially accepted these comments. The staff applied these comments generically to the staff’s updates of the SRP-SLR Table 3.1-1, items 118 and 119 in Appendix A of the ISG and to the staff’s updates of all “R-423” and “R-424” type line items in GALL-SLR Report Tables IV.B2, IV.B3, and IV.B4, as cited and revised in Appendices B.1, B.2, and B.3 of the ISG. Upon further review, the staff agrees with the EPRI MRP that further changes can be made to SRP-SLR items 118 and 119 and the GALL-SLR Report type “R-423” and “R-424” items for consistency objectives, but not necessarily in the exact manner that EPRI recommended in this set of comments. Since the set of components in a specified GALL-SLR Report item is commonly only a subset of the components listed in a related SRP-SLR item, the component descriptions in the GALL-SLR Report “R-423” and “R-424” items only need to be similar to the component descriptions for the analogous items in SRP-SLR Table 3.1-1, which is the correlation of SRP-SLR item 3.1-1-118 to the “R-423” items for cracking mechanisms and the correlation of SRP-SLR item 3.1-1-119 to the “R-424” items for non-cracking mechanisms. Therefore, the staff updated the “Component” column entries of SRP-SLR Table 3.1-1, items 118 and 119, to state: “*Stainless steel, nickel alloy . . . PWR reactor vessel internals components or LRA/SLRA-specified reactor vessel internal component exposed to reactor coolant, neutron flux*”. Stellite was also included in the adjustment of the component description in SRP-SLR item 3.1-1-119. The staff also updated the “Structure and/or Component” column entries GALL-SLR Report items IV.B2.R-R-423, IV.B3.R-423, IV.B4.R-423, IV.B2.R-424, IV.B3.R-424, and IV.B4.R-424 to state: “*Reactor vessel internal components or LRA/SLRA specified reactor vessel internal component*”. The staff updated the “Aging Management Program (AMP)/TLAA” column entries of the “R-423” items to be consistent with the corresponding column entry in SRP-SLR item 3.1-1-118, and updated the “Aging Management Program (AMP)/TLAA” column entries of the “R-424” items to be consistent with the

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EPRI MRP #14, #15, #16, #17, #22 and #23 (Cont.)			corresponding column entry in SRP-SLR item 3.1-1-119. The staff also updated the “Material” column entries of the “R-424” items to state: “*Stainless steel, nickel alloy, stellite (as a wear-resistant surface)*”. Stellite needed to be added as part of the final adjustment of the component description of SRP-SLR item 3.1-1-119 and the material descriptions in the GALL-SLR Report “R-424” items in order to be consist with industry comments that stellite has been used as a material wear-resistant surface for some RVI components.
EPRI MRP #18	ISG Appendix B.1, GALL-SLR Table IV.B2, Item IV.B2.RP-302a (Page 15 of the appendix)	EPRI wrote the following as a basis for requesting deletion of loss of fracture toughness due to neutron irradiation embrittlement and loss of preload due to irradiation-enhanced stress relaxation or creep as additional cited non-cracking effect and mechanism combinations for GALL-SLR Item IV.B2.RP-302a in ISG Appendix B.1: “The irradiation effects added for this component appear to originate from MRP-191, Revision 1. They show up in that revision, but the more refined analysis of MRP-191, Revision 2 showed that these irradiation effects were not a concern for the thermal shield flexures.”	The staff accepted EPRI MRP Comment #18 and the EPRI MRP’s statement that the update of GALL-SLR Report item IV.B2.RP-302a in the ISG should not include additional cited irradiation effects for Westinghouse-design thermal shield flexures. Specifically, based on lessons learned from the staff’s review of the Surry SLRA gap analysis, the past applicant indicated that the thermal shield flexures for the Surry units did not screen in for void swelling (VS), irradiation embrittlement (IE),or irradiation-enhanced stress relaxation or irradiation-enhanced creep (ISR/IC) per EPRI’s 80-Year Expert Panel analysis of the components in MRP-2018-022. Specifically, the gap analysis indicates that the projected 80-year fluence exposures of the thermal shield flexures are in a fluence zone lower than the threshold for screening the thermal shield flexures in for the referenced irradiation mechanisms. Since this ISG only reports changes to the existing guidance for PWR RVI components in the GALL-SLR Report and SRP-SLR, and since the staff has determined as a result of this comment that there is no change to GALL-SLR Report item IV.B2.RP-302a, the staff removed this item from the final version of this ISG. The item in the GALL-SLR Report stands unrevised.

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EPRI MRP #19	ISG Appendix B.2, GALL-SLR Table IV.B3, Item IV.B3.RP-320 (Page 2 of the appendix)	EPRI wrote the following regarding the staff's basis for adding CE design upper internals assembly guide lug inserts to the draft update of GALL-SLR Item IV.B3.RP-320 in ISG Appendix B.2: "It is unclear why the guide lug inserts were added to this line for cracking effects. They are only screened in for wear and IE in MRP-191, Rev. 2."	The staff did not accept Comment #19. Based on the staff's rejection of NEI Comment #1, the staff cannot use MRP-191, Rev. 2 as a basis for citing component-specific aging effects or mechanisms in this ISG. Furthermore, in MRP-227, Rev. 1-A, Table 3-2, the guide lugs and guide lug inserts and bolts, are screened in as Existing Program components (i.e., "X" designations) for the mechanisms of fatigue, wear, and irradiation-enhanced stress relaxation or creep (ISR/IC), even though the EPRI MRP did not specifically reflect fatigue as a cited mechanism in the Existing Program line items for the components in Line Items C13 and C14 in Table 4-8 of the MRP-227, Rev. 1-A. In Footnote 3 of Table 3-2 in MRP-227, Rev. 1-A, EPRI makes the following statement relative to potential degradations that may occur in the guide lug fixtures: "*Bolt deterioration may lead to degradation in the lug fixtures. Inspection recommendations relate to the entire guide lug fixture.*" Thus, the inclusion and updated version of GALL-SLR Report item IV.B3.RP-320 in Appendix B.2 of the draft ISG remains valid for the final version of the line item in the final ISG and it is appropriate for the staff to include the guide lug inserts in the scope of the line item. Based on Footnote 3 in Table 3-2 of MRP-227, Rev. 1-A, and for simplicity of the "RP-320" line item, the staff conservatively applied fatigue to all of the specified guide lug fixture components that are within the scope of the "RP-320" item. The staff assumes the ASME Section XI VT-3 examinations credited for the guide lug fixtures in Table 4-8 of MRP-227, Rev. 1-A, are sufficient to detect any potential cracking or wear that may occur in the fixtures, and additionally, any loss of preload that may occur in the lug bolts. However, there is no requirement that forces an applicant to use the "RP-320" item for its SLRA if the IPA concludes that cracking is not any aging effect that requires management for the specified guide luge fixture components. The applicant could also apply the "RP-320" item as a consistent-with-GALL item for only some of the referenced guide lug fixture components in the line item if that is consistent with the IPA basis.

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EPRI MRP #20 and #21	ISG Appendix B.2, GALL-SLR Table IV.B3, Items IV.B3.RP-333 and IV.B3.RP-333a (Pages 8 and 9 of the appendix)	EPRI wrote the following regarding the staff's basis for adding IASCC as a listed aging mechanism for CE design core support barrel lower flange welds (LFWs) in the staff's draft updates of GALL-SLR Items IV.B3.RP-333 and for deleting GALL-SLR item IV.B3.RP-333a on the topic of loss of material due to neutron irradiation embrittlement in the LFWs: For Comment #20 "Like with overall comments 11 and 12, IASCC was included in MRP-191, Rev. 2 because the LGW/LFW and MGW were combined on the same line. IASCC is not expected at the elevation of the LGW/LFW." For Comment #21 made in relation to IE of the LFWs, "Similar discussion to overall comments 11, 12, and 20."	The staff did not accept EPRI MRP Comments #20 and #21 or the EPRI MRP's collective statements in the two comments that neither irradiation-assisted stress corrosion cracking (IASCC) nor irradiation embrittlement (IE) should be attributed to CE design core support barrel lower girth welds (LGWs)/lower flange welds (LFWs). In EPRI MRP's response to Request for Additional Information (RAI) #26, Item "a" on EPRI Report MRP-227, Rev. 1 (as submitted in EPRI Letter No. MRP 2017-27 dated October 16, 2017, ADAMS Accession No. ML17305A056), EPRI identified that IASCC and IE are applicable irradiation mechanisms for CE-design LGWs/LFWs. Although the EPRI MRP may have performed further studies to exclude IASCC and IE as applicable mechanisms for CE design core support barrel LGWs/LFWs in the upcoming MRP-227, Rev.2, report, the report has yet to docketed with the NRC or reviewed or accepted by the staff. Thus, for the final issuance of the ISG, IASCC will remain as a cited irradiation-based cracking mechanism for CE-design core support barrel LGWs/LFWs in the ISG's update of GALL-SLR Report item IV.B3.RP-333 and IE will remain as a cited non-cracking, irradiation-based aging mechanism for CE-design core support barrel LGWs/LFWs in the ISG's update of GALL- SLR Report item IV.B3.RP-333a.

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EPRI MRP #24, #30, and #33	For #24, ISG Appendix B.4, GALL-SLR Table IV.E, Item IV.E.R-44 (Page 2 of ISG Appendix B.4) For #30, ISG Appendix A, SRP-SLR Table 3.1-1, Items 114, 118, and 119 (Pages 10 and 11 of ISG Appendix A) For #33, ISG Appendix B.3, various AMR Items IV.B4.R-423 (as referenced to SRP-SLR Item 118) and IV.B4.R-424 (as referenced to SRP-SLR Item 119) (Pages 12 and 13 in ISG Appendix B.3)	EPRI made the following editorial comments in regard to the staff's update of GALL-SLR Report item IV.E.R-44 in Appendix B.4 of the ISG and updates of SRP-SLR Table 3.1-1, items 114, 118, and 119 in Appendix A of the ISG: "Should this line be marked as "M" for modified rather than "N"? Change to M."	The staff did not accept EPRI Comments #24, #30, and #33. The staff defines and discusses its criteria for "New" (N), "Modified" (M), Edited" (E), or "Deleted" (D) AMR item designations in Section 1.2 of NUREG-2192 (the SRP-SLR). For comparisons to the "Table 1" AMR summary items for PWR RVI components in Table 3.1-1 (as updated in Appendix A of the ISG), the "N", "M", "E", or "D" designations for AMR summary items are made in comparison to the versions of the AMR summary items in Table 3.1-1 in the prior license renewal SRP, (i.e., SRP-LR, Rev. 2), not in comparison to the version of these items in the SRP-SLR Report. Similarly, for comparisons to the "Table 2" type AMR items for PWR RVI components in Tables IV.B2, IV.B3, IV.B4, and IV.E (as updated in Appendices B.1, B.2, B.3 and B.4 of the ISG, respectively), the "N", "M", "E", or "D" designations for AMR items are made in comparison to the versions of the AMR items in Tables IV.B2, IV.B3, IV.B4, and IV.E in the previous license renewal GALL Report (i.e., GALL Report, Rev. 2), and not the versions of these AMR items in the corresponding tables of the GALL-SLR Report.

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EPRI MRP #25	ISG Appendix D, GALL-SLR AMP XI.M16A, 'PWR Vessel Internals" – Program Description and Program Elements in the AMP	EPRI made the following editorial comment in regard to MRP-227 references in GALL AMP XI.M16A, "PWR Vessel Internals," as updated in ISG Appendix D: "The references to MRP-227 are mixed between "MRP-227, Revision 1-A" and "MRP-227 (as supplemented)". In multiple cases, it seems like this should be the second one."	The staff accepted EPRI MRP Comment #25. Specifically, the staff performed a review of the individual MRP-227, Revision 1-A references in the update of GALL-SLR Report AMP XI.M16A, "PWR Vessel Internals," in Appendix D of the ISG versus those stating "MRP-227 (as supplemented)" in the updated version of the AMP. The staff found at least one case where the referenced MRP-227 terminology in the updated version of GALL-SLR AMP XI.M16A, "PWR Vessel Internals," should be switched to either "MRP-227, Rev. 1-A" or "MRP-227 (as supplemented)" consistent with the comment statement. But to clarify, the AMP program description and program element criteria reference MRP-227, Revision 1-A directly if the context of the sentence is referring to criteria in MRP-227, Rev. 1-A when used as a starting point for the AMP. In contrast, the revisions of the AMP specify "MRP-227 (as supplemented)" if the I&E criteria in MRP-227, Rev. 1-A for a given component are being amended or supplemented by criteria in supplemental methodologies. Examples of the latter case are the citing and use of MRP-2018-022 for the gap analysis of the past Turkey Point and Surry SLRAs, or the past citing in the Surry SLRA of MRP-2019-023 as a supplemental one-time inspection protocol for the core barrel middle and lower axials welds in the units. For implementation criteria, the staff should be referencing Chapter 7 of the MRP-227, Rev. 1-A because that section establishes the most up-to-date staff-approved implementation criteria for these types of living programs. The staff's understanding is that Chapter 7 in MRP-227, Rev. 1-A would allow supplemental methodologies to be incorporated and used by the licensee for the "living" programs on an as-needed basis. The staff has updated the MRP-227 terminology references in the updated version of the AMP accordingly.

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EPRI MRP #26 and #27	For #26, ISG Appendix D, GALL-SLR AMP XI.M16A, ‘PWR Vessel Internals,” References Section (Page 8 of the appendix) For #27, ISG Appendix D, GALL-SLR Table X-01 FSAR Supplement Example for GALL-SLR AMP XI.M16A (Page 10 of the appendix)	EPRI made the following comments for these AMP sections in ISG Appendix D: “MRP-228 is currently at Revision 3. Update reference to Revision 3.”	The staff accepted Comments #26 and #27. The staff has updated the MRP-228 references in the staff’s update of GALL-SLR Report AMP XI.M16A, “PWR Vessel Internals,” and FSAR Supplement Example for the AMP in Appendix D of the ISG to cite Revision 3 of the MRP-228 report. This includes the staff’s correction of the MRP-228 reference in the References section of AMP XI.M16A to be that for MRP-228, Rev. 3 report, and to specify it as EPRI Proprietary Topical Report No. 3002010399, “Materials Reliability Program: Inspection Standard for PWR Internals (MRP-228, Rev. 3)” (Non-publicly available in ADAMS Accession No. ML19081A064), November 2018. A non-proprietary version of the MRP-228, Revision 3 report may be accessed by members of the public at ADAMS Accession No. ML19081A058.
EPRI MRP #28	ISG and ISG Appendices	EPRI made the following generic comment for the contents of the ISG: " ‘EPRI Technical Report’ was changed to ‘EPRI Topical Report’ in two locations in Appendix D, but not changed in several other locations within the document. Change ‘EPRI Technical Report’ to ‘EPRI Topical Report’.”	The staff accepted EPRI MRP Comment #28. The staff found five instances in the ISG where the words “EPRI Technical Report” should be replaced with the words “EPRI Topical Report.” The staff has made the appropriate adjustments of the terminology in the ISG consistent with EPRI MRP’s request in Comment #28.

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EPRI MRP #29 and #32	ISG Appendix B.3, Various AMR line items for GALL-SLR Table IV.B4 in the appendix ISG Appendix A, and Various AMR line items for SRP-SLR Table 3.1-1 in the appendix.	EPRI made the following generic comments for the contents of the updated AMR line items for GALL-SLR Report Table IV.B4 in Appendix B.3 and SRP-SLR Table 3.1-1 in Appendix A of the ISG: Comment 29: “There are several 'GALL-SLR Item' entries in this table for the B&W units (IV.B4.RP…) that seem to identify aging effects/mechanisms that are not consistent with MRP-227, Rev. 1-A. For example, GALL-SLR Item IV.B4.RP-242a is an entry for the CRGT spacer castings (ID 51a). Per MRP-227, Rev. 1-A, the CRGT spacer castings are only potentially susceptible to TE, but the Table 3.1-1 entry for IV.B4.RP-242a (ID 51a) states that the 'aging effect/mechanism' is 'cracking due to SCC, IASCC, fatigue'. As this is Objective 1 for this SLR ISG, review the entries in Table 3.1-1 to ensure they identify aging effects/mechanisms consistent with MRP-227, Rev. 1-A.” Comment 32: “IV Table B4 provides various items and lists their applicable aging effect/mechanism in one of the columns in the table. It appears there are some aging effects/mechanisms that are not consistent with MRP-227, Rev. 1-A in several table entries. For example, item IV.B4.RP-249a (baffle plates) identifies the aging effect/mechanism of 'cracking due to IASCC or fatigue'. Per MRP-227, Rev. 1-A, the baffle plates are only potentially susceptible to IE.”	EPRI MRP Comments #29 and #32 are somewhat analogous to NEI’s comment made in NEI Comment #3. The staff partially accepts the comments and rationale made in EPRI MRP Comments #29 and #32. Consistent with the staff’s basis for responding to NEI Comment #3, the staff does not always use aging effect and mechanism bases in the MRP-277, Revision 1-A report as the sole basis for establishing the aging effects and mechanisms for referenced SRP-SLR AMR line items that were updated in Appendix A of the ISG or GALL-SLR Report AMR line items updated in Appendices B.1, B.2, B.3 or B.4 of the ISG. Unlike the staff’s review of the AMR line items for Westinghouse-designed PWR RVI components in which the staff used lessons learned from previous SLRA reviews, the staff does not have any current 80-year lessons learned criteria that the staff can apply to the assessment of the AMR line items for B&W-designed RVI components, as the staff has yet to approve any docketed SLRAs for B&W PWRs. For this reason, the staff adjusted the AMR items for B&W designed components to reflect the guidance in MRP-227, Rev. 1-A. For the example in Comment #29 regarding GALL-SLR Report item IV.B4.RP-242a, the adjustment reverted the item to what is listed in table IV.B4 of the original GALL-SLR Report. Therefore, these items and other B&W AMR items that were adjusted in this final ISG to reflect the original GALL-SLR Report have been removed from the final version of this ISG. For the example in Comment #32 regarding GALL-SLR Report items IV.B4.RP-249a, the adjustment resulted in deletion of the AMR item from the GALL-SLR Report, as reflected in this final ISG.

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EPRI MRP #31	ISG Appendix B.3, GALL-SLR Table IV.B4, Items IV.B4.RP-247 and IV.B4.RP-247c (Page 2 of the ISG appendix)	EPRI provided the following comment as the for modifying the listed materials in GALL-SLR Item IV.B4.RP-247 and IV.B4.RP-247c. “ONS-1 has 12 additional LCB bolts, which are fabricated from Type 304 stainless steel (see ML003708443, pages 2-12 and 2-23). These entries in IV Table B4 for the LCB bolts state the material is "stainless steel, nickel alloy". As the additional 12 Type 304 LCB bolts at ONS-1 are not included in MRP-227, Rev. 1-A for examination (see Table 3-1, page 3-21 where LCB bolts are specifically identified as Alloy A-286 or Alloy X-750), this material description of 'stainless steel' could be confusing and/or misinterpreted.”	The staff did not accept the comment statements made by EPRI MRP in EPRI MRP Comment #31. The staff acknowledges that the line item entry for B&W-design lower core barrel (LCB) bolts in Table 3-1 of MRP-227, Rev. 1-A reports cites the bolt material as being an A-286 type stainless steel or X-750 nickel alloy material. However, the staff has decided to list the more generic stainless steel and Nickel alloy material designations for B&W design lower core barrel (LCB) bolts in GALL-SLR Report items IV.B4.RP-247 and IV.B4.RP-247c, or other types of B&W-design bolts that EPRI had indicated are made from the specific materials (e.g., UCB bolts covered by item IV.B4.RP-248, etc.). The first reason for citing the more generic classifications of the materials is that the Type A-286 stainless steel materials do not have an individual line item in GALL-SLR Report Table IX.C, “Materials,” and are instead a subset of the “stainless steel” material definition in the table; similarly, Type X-750 materials do not have an individual material definition in GALL-SLR Report Table IX.C and are identified as a subset of the generic “Nickel alloys” material definition in the table. Secondly, the AMR item for inspecting the LCB bolts and their bolt locking devices is provided in Item B8 in Table 4-1 of the MRP-227, Rev. 1-A report. Item B8 includes Note 6 on the line item, which states: “*A minimum of 75% of the total population (examined + unexamined), including coverage consistent with the Expansion criteria in Section 5.3 of this document, must be examined for inspection credit.”* The footnote on the B8 line item does not define ”total population” in terms of a specified bolt material type. In addition, the AMR line item criteria in the GALL-SLR report are included only for potential LRA or SLRA AMR identification and application objectives and do not have any bearing on how an applicant or licensee would interpret the MRP-227, Rev. 1-A inspection or evaluation (I&E) guidelines under the scope of the licensee’s PWR Vessel Internals Program. Thus, for the LCB bolt example, it is the licensee’s responsibility to determine what is appropriate for LCB bolt inspections in order to achieve EPRI’s specified 75% bolt population criterion under Item B8 of Table 4-1 in MRP-227, Rev. 1-A report. The same logic applies to other B&W bolt types.