FINAL LICENSE RENEWAL INTERIM STAFF GUIDANCE

LR-ISG-2011-04

UPDATED AGING MANAGEMENT CRITERIA FOR REACTOR VESSEL INTERNAL COMPONENTS FOR PRESSURIZED WATER REACTORS

INTRODUCTION

This license renewal interim staff guidance (LR-ISG) updates the U.S. Nuclear Regulatory Commission (NRC's) guidance in NUREG-1800, Revision 2, "Standard Review Plan for Review of License Renewal Applications for Nuclear Power Plants," (SRP-LR) and NUREG-1801, Revision 2, "Generic Aging Lessons Learned Report" (GALL Report). This LR-ISG is primarily based on the issuance of Revision 1 to the Final Safety Evaluation (SE) 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," by letter dated December 16, 2011 (SE, Revision 1, on MRP-227) (ADAMS Accession No. ML11308A770). After the issuance of the staff's SE, Revision 1, on MRP-227, EPRI Technical Report No. 1022863 (MRP-227-A) was published in January 2012. MRP-227-A is the NRC-endorsed version of MRP-227, which incorporates the NRC staff's SE, Revision 1, on MRP-227. Specifically, this LR-ISG revises the recommendations in the GALL Report and the NRC staff's acceptance criteria and review procedures in the SRP-LR to ensure consistency with MRP-227-A. This LR-ISG also provides a framework to ensure that PWR license renewal applicants will adequately address age-related degradation and aging management of reactor vessel internal (RVI) components during the term of the renewed license.

DISCUSSION

Current Regulatory Framework

Pursuant to Part 54, "Requirements for Renewal of Operating Licenses for Nuclear Power Plants," Section 21(a)(3), of Title 10 of the *Code of Federal Regulations* (10 CFR 54.21(a)(3)), a license renewal applicant is required to perform an integrated plant assessment (IPA) that demonstrates that the effects of aging on structures and components subject to an aging management review (AMR) are adequately managed so that the intended function(s) will be maintained consistent with the current licensing basis (CLB) for the period of extended operation. The NRC's guidance in SRP-LR Section 3.0.1 defines the AMR as the identification of the structure and component materials, environments, aging effects, and aging management programs (AMPs) credited for managing the aging effects. In turn, SRP-LR Section A.1.2.3 defines an acceptable AMP as consisting of 10 elements. In addition, 10 CFR 54.21(d) requires the license renewal application (LRA) to contain a final safety analysis report (FSAR) supplement that includes a summary description of the programs and activities for managing the effects of aging and the evaluation of time-limited aging analyses (TLAAs) for the period of extended operation.

GALL Report AMP XI.M16A, "PWR Vessel Internals," provides recommendations for an AMP to manage the effects of aging for PWR RVI components. In addition, the GALL Report provides component-specific AMR items for PWR RVI components in the following tables:

- Table IV.B2 for Westinghouse-designed RVI components
- Table IV.B3 for CE-designed RVI components
- Table IV.B4 for B&W-designed RVI components

SRP-LR Table 3.1-1 provides the specific commodity group-based AMR items for PWR RVI components. SRP-LR Sections 3.1.2.2.1, 3.1.2.2.3, 3.1.2.2.9, 3.1.2.2.10, 3.1.2.2.12, 3.1.2.2.13, and 3.1.2.2.14 provide the aging management review results for which further evaluation is recommended by the GALL Report for PWR RVI components. Finally, SRP-LR Table 3.0-1 provides an example of the type of information to be included in the FSAR Supplement for an AMP for PWR RVI components.

Basis for Issuing Interim Guidance

On January 12, 2009, EPRI submitted Technical Report No. 1016596, "Materials Reliability Program: Pressurized Water Reactor Internals Inspection and Evaluation Guidelines (MRP-227, Revision 0)," for NRC staff review and approval. On June 22, 2011, the NRC staff issued its SE on MRP-227, Revision 0, which contained specific topical report condition items (TRCIs) on the use of MRP-227 and Applicant/Licensee Action Items (A/LAIs) that must be addressed by those applicants or licensees utilizing this topical report. The staff issued a revision of its SE on the report methodology (i.e., SE, Revision 1, on MRP-227) by letter dated December 16, 2011. MRP-227-A, the NRC-endorsed version of MRP-227, was later published in January 2012 and provides guidance for a PWR licensee or license renewal applicant to use in the development and implementation of an AMP for RVI components. MRP-227-A also incorporates A/LAIs that are to be addressed if this report is referenced to satisfy the requirements of 10 CFR 54.21(a)(3) for demonstrating that the effects of aging on the RVI components, within the scope of MRP-227, will be adequately managed. The staff recommends that a PWR license renewal applicant provide its responses to these A/LAIs in Appendix C of the LRA. The use of MRP-227-A by a PWR license renewal applicant is not a substitute for performing a plant-specific IPA to identify those structures and components subject to an aging management review, in accordance with 10 CFR 54.21(a)(1).

Regulatory Issue Summary (RIS) 2011-07, "License Renewal Submittal Information For Pressurized Water Reactor Internals Aging Management," dated July 21, 2011, was issued, in part, to facilitate a predictable and consistent method for reviewing the aging management of RVI components for commercial PWR LRAs. An "inspection plan" is one aspect of the A/LAIs of the staff's SE, Revision 1, for MRP-227. This "inspection plan" provides information about the RVI components to be inspected and a description of how they will be managed for age-related degradation. Details of an "inspection plan" for those PWR plant licensees that have not submitted but plan to submit an LRA in the future will be incorporated into the LRA as part of the 10-element aging management program and aging management review line items. Thus, consistent with RIS 2011-07, these future license renewal applicants need not submit a separate document that contains an "inspection plan" in response to the A/LAIs of the staff's SE, Revision 1, for MRP-227.

Prior to the completion of its review and issuance of the SE on MRP-227, the staff issued SRP-LR, Revision 2, and GALL Report, Revision 2, in December 2010. Since SRP-LR, Revision 2, and GALL Report, Revision 2, were based on MRP-227, Revision 0, the relevant

portions of the SRP-LR, Revision 2, and GALL Report, Revision 2, are now being updated with this LR-ISG to reconcile any differences with MRP-227-A.

ACTION

This LR-ISG updates the GALL Report, Revision 2, and SRP-LR, Revision 2, to ensure consistency with MRP-227-A for the aging management of age-related degradation for PWR RVI components during the term of a renewed operating license. Appendix A, "Revisions to the GALL Report and SRP-LR," to this LR-ISG shows these changes. The majority of these changes result in the incorporation of MRP-227-A within the SRP-LR, Revision 2, and the GALL Report, Revision 2. To better show these changes, a mark-up is shown in Appendix B, "Mark-Up of Changes to the GALL Report and SRP-LR," to this LR-ISG.

On March 20, 2012, at Volume 77, page 16270, of the Federal Register (77 FR 16270), the NRC requested public comments on draft LR-ISG-2011-04. Subsequently, as noticed on April 19, 2012, at Volume 77, page 23513, of the Federal Register (77 FR 23513), the NRC issued an editorial correction to the original notice to specifically identify the ADAMS Accession Nos. for additional documents associated with draft LR-ISG-2011-04.

The staff received comments on draft LR-ISG-2011-04 by letters from EPRI and the Pressurized Water Reactor Owners Group Materials Subcommittee (ADAMS Accession No. ML12146A267) and from the Nuclear Energy Institute (ADAMS Accession No. ML12144A147). The staff considered all comments, and its evaluation of these comments is contained in Appendix C, "Staff Response to Public Comments on Draft License Renewal Interim Staff Guidance 2011-04," of this LR-ISG. The guidance described in this final LR-ISG supersedes the affected sections of the SRP-LR, Revision 2, and the GALL Report, Revision 2, and is approved for use by the NRC staff and stakeholders.

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

Any structures and components identified in this LR-ISG as requiring aging management, which were not previously identified in earlier versions of the SRP-LR, Revision 2, or GALL Report, Revision 2, are considered by the staff to be newly-identified structures and components under 10 CFR 54.37(b).

BACKFITTING AND ISSUE FINALITY

This LR-ISG contains guidance on one acceptable approach for managing the associated aging effects during the PEO for components within the scope of license renewal. The staff's discussion on compliance with the requirements of the Backfit Rule, 10 CFR 50.109 is presented below.

Compliance with the Backfit Rule and Issue Finality

Issuance of this LR-ISG does not constitute backfitting as defined in 10 CFR 50.109(a)(1), and the NRC staff did not prepare a backfit analysis for issuing this LR-ISG. There are several rationales for this conclusion, depending on the status of the nuclear power plant licensee.

Licensees currently in the license renewal process – The backfitting provisions in 10 CFR 50.109 do not protect an applicant, as backfitting policy considerations are not applicable to an

applicant. Therefore, issuance of this LR-ISG does not constitute backfitting as defined in 10 CFR 50.109(a)(1). There currently are no combined licenses (i.e., 10 CFR Part 52) license renewal applicants; therefore, the changes and new positions presented in the LR-ISG may be made without consideration of the issue finality provisions in 10 CFR Part 52, "Licenses, Certifications, and Approvals for Nuclear Power Plants."

Licensees who already hold a renewed license – This guidance is nonbinding and the LR-ISG would not require current holders of renewed licenses to take any action (i.e., programmatic or plant hardware changes for managing the associated aging effects for components within the scope of this LR-ISG). Current holders of renewed licenses should treat this guidance as operating experience and take actions as appropriate to ensure that applicable aging management programs are, and will remain, effective. If, in the future, the NRC decides to take additional action and impose requirements for managing the associated aging effects for components within the scope of this LR-ISG, then the NRC would follow the requirements of the Backfit Rule.

Current operating license or combined license holders who have not yet applied for renewed licenses – The backfitting provisions in 10 CFR 50.109 do not protect any future applicant, as backfitting policy considerations are not applicable to a future applicant. Therefore, issuance of this LR-ISG does not constitute backfitting as defined in 10 CFR 50.109(a)(1). The issue finality provisions of 10 CFR Part 52 do not extend to the aging management matters covered by 10 CFR Part 54, as evidenced by the requirement in 10 CFR 52.107, "Application for Renewal," stating that applications for renewal of a combined license must be in accordance with 10 CFR Part 54.

APPENDICES

Appendix A provides the staff's revisions to the SRP-LR, Revision 2, and the GALL Report, Revision 2, for managing aging in PWR RVI components and includes the following sections:

- Section 1 – Revised version of the GALL Report
- Section 2 – Revised version of the SRP-LR

Appendix B provides a mark-up of the SRP-LR, Revision 2, and GALL Report, Revision 2, to better show the changes made as a result of LR-ISG-2011-04 and includes the following sections:

- Section 1 – Mark-up of changes to the GALL Report
- Section 2 – Mark-up of changes to the SRP-LR

Appendix C provides the staff's bases for resolving comments that were received on the draft LR-ISG-2011-04.

REFERENCES

1. *U.S. Code of Federal Regulations*, "Domestic Licensing of Production and Utilization Facilities," Part 50, Chapter I, Title 10, "Energy."

2. *U.S. Code of Federal Regulations*, "Licenses, Certifications, and Approvals for Nuclear Power Plants," Part 52, Chapter I, Title 10, "Energy."

3. *U.S. Code of Federal Regulations*, "Requirements for Renewal of Operating Licenses for Nuclear Power Plants," Part 54, Chapter I, Title 10, "Energy."

4. U.S. Nuclear Regulatory Commission, "Generic Aging Lessons Learned (GALL) Report," NUREG-1801, Revision 2, December 2010, ADAMS Accession No. ML103490041.

5. U.S. Nuclear Regulatory Commission, "Standard Review Plan for Review of License Renewal Applications for Nuclear Power Plants," NUREG-1800, Revision 2, December 2010, ADAMS Accession No. ML103490036.

6. U.S. Nuclear Regulatory Commission, Final Safety Evaluation of EPRI Report, Materials Reliability Program Report 1016596 (MRP-227), Revision 0, *Pressurized Water Reactor Internals Inspection and Evaluation Guidelines,* June 22, 2011, ADAMS Accession No. ML111600498.

7. U.S. Nuclear Regulatory Commission, Revision 1 to the Final Safety Evaluation of Electric Power Research Institute (EPRI) Report, Materials R*eliability Program (MRP) Report 1016596 (MRP-227), Revision 0, Pressurized Water Reactor Internals Inspection and Evaluation Guidelines*, December 16, 2011, ADAMS Accession No. ML11308A770.

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

9. Electric Power Research Institute, EPRI Technical Report No. 1022863, Materials Reliability Program: Pressurized Water Reactor Internals Inspection and Evaluation Guidelines (MRP-227-A), December 2011, ADAMS Accession No. ML12017A193 (Transmittal letter from the EPRI-MRP) and ADAMS Accession Nos. ML12017A194, ML12017A196, ML12017A197, ML12017A191, ML12017A192, ML12017A195 and ML12017A199 (Final Report).

10. U.S. Nuclear Regulatory Commission, "Updated Aging Management Criteria for Reactor Vessel Internal Components of Pressurized Water Reactors," Federal Register, Vol. 77, No. 54, March 20, 2012, pp. 16270-16271.

11. U.S. Nuclear Regulatory Commission, "Updated Aging Management Criteria for Reactor Vessel Internal Components of Pressurized Water Reactors," Federal Register, Vol. 77, No. 76, April 19, 2012, pp. 23513.

12. T. Wells and E. Fernandez, Electric Power Research Institute Materials Reliability Program and the Pressurized Water Reactor Owners Group Materials Subcommittee, letter to Document Control Desk, U.S. Nuclear Regulatory Commission, May 21, 2012, ADAMS Accession No. ML12146A267.

13. M. Richter, Nuclear Energy Institute, letter to Cindy K. Bladey, U.S. Nuclear Regulatory Commission, May 21, 2012, ADAMS Accession No. ML12144A147.

14. U.S. Nuclear Regulatory Commission, Nuclear Regulatory Commission Regulatory Issue Summary 2011-07, *License Renewal Submittal Information For Pressurized Water*

Reactor Internals Aging Management, July 21, 2011, ADAMS Accession No. ML111990086.

15. U.S. Nuclear Regulatory Commission. 2008. Memorandum from Dale E. Klein, Chairman, to Hubert T. Bell, Office of the Inspector General, "Response to Recommendation 8 of 9/6/07 Audit Report on NRC's License Renewal Program." (April 1, 2008). ADAMS Accession No. ML080870286.

Appendix A

REVISIONS TO THE GALL REPORT AND SRP-LR

Appendix A, Section 1 – Revised version of the GALL Report

(1) Revised version of GALL Report AMP XI.M16A

XI.M16A PWR VESSEL INTERNALS

Program Description

This program relies on implementation of the Electric Power Research Institute (EPRI) Technical Report No. 1022863, "Materials Reliability Program: Pressurized Water Reactor (PWR) Internals Inspection and Evaluation Guidelines," (MRP-227-A) and EPRI Technical Report No. 1016609, "Materials Reliability Program: Inspection Standard for PWR Internals," (MRP-228) to manage the aging effects on the pressurized water reactor (PWR) reactor vessel internal (RVI) components. The recommended activities in MRP-227-A and additional plant-specific activities not defined in MRP-227-A are implemented in accordance with Nuclear Energy Institute (NEI) 03-08, "Guideline for the Management of Materials Issues." The staff approved the augmented inspection and evaluation (I&E) criteria for PWR RVI components in NRC Safety Evaluation (SE), Revision 1, on MRP-227 by letter dated December 16, 2011.

This program is used to manage the effects of age-related degradation mechanisms that are applicable in general to the PWR RVI components at the facility. These aging effects include: (a) cracking, including stress corrosion cracking (SCC), primary water stress corrosion cracking (PWSCC), irradiation-assisted stress corrosion cracking (IASCC), and cracking due to fatigue/cyclic loading; (b) loss of material induced by wear; (c) loss of fracture toughness due to either thermal aging or 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.

The program applies the guidance in MRP-227-A 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.

MRP-227-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 PWR designs were assigned to one of the following four groups: "Primary," "Expansion," "Existing Programs," and "No Additional Measures." Definitions of each group are provided in "Generic Aging Lessons Learned Report" (GALL Report), Revision 2, Chapter IX.B.

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 of "Expansion" internals component locations that are specified to expand the sample should the indications be more severe than anticipated.

The degradation effects in a third set of internals locations are deemed to be adequately managed by "Existing Programs," such as American Society of Mechanical Engineers

(ASME) Code, Section XI,[11] Examination Category B-N-3, examinations of core support structures. A fourth set of internals locations are deemed to require "No Additional Measures."

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 in MRP-227-A, which provides an augmented inspection and flaw evaluation methodology for assuring the functional integrity of safety-related internals in commercial operating U.S. PWR nuclear power plants designed by Babcock & Wilcox (B&W), Combustion Engineering (CE), and Westinghouse. The scope of components considered for inspection in MRP-227-A includes core support structures, those RVI components that serve an intended license renewal safety function pursuant to criteria in 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.

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

2. *Preventive Actions:* MRP-227-A 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 AMP XI.M2, "Water Chemistry."

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

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

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.

4. *Detection of Aging Effects:* The inspection methods are defined and established in Section 4 of MRP-227-A. 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 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.

In some cases (as defined in MRP-227-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 MRP-227.

5. *Monitoring and Trending:* The methods for monitoring, recording, evaluating, and trending the data that result from the program's inspections are given in Section 6 of MRP-227-A and its subsections. 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. The

examination and re-examinations that are implemented in accordance with MRP-227-A, together with the criteria specified in MRP-228 for inspection methodologies, 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 a 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 to the applicable inspection technique and the resulting impact on the intended function(s) of the assembly containing the components.

6. *Acceptance Criteria:* Section 5 of MRP-227-A, which includes Table 5-1 for B&W-designed RVIs, Table 5-2 for CE-designed RVIs, and Table 5-3 for Westinghouse-designed RVIs, provides the specific examination and flaw evaluation acceptance criteria for the “Primary” and “Expansion” RVI component examination methods. 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.

7. *Corrective Actions:* 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, 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:* 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 MRP-227-A, in conjunction with NEI 03-08 and other guidance documents, reports, or methodologies 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:* 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 evaluation in Section 3.5 of the NRC's SE, Revision 1, on MRP-227 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 for its impacts on the program, including implementing procedures, are governed by NEI 03-08 and Appendix A of MRP-227-A. Consistent with MRP-227-A, the reporting of inspection results and operating experience 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 operating experience, as discussed in Appendix B of the GALL Report, which is documented in LR-ISG-2011-05.

References

10 CFR Part 50, Appendix B, *Quality Assurance Criteria for Nuclear Power Plants*, Office of the Federal Register, National Archives and Records Administration, 2011.

10 CFR Part 50.55a, *Codes and Standards*, Office of the Federal Register, National Archives and Records Administration, 2011.

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

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

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

EPRI Technical Report No.1022863, *Materials Reliability Program: Pressurized Water Reactor Internals Inspection and Evaluation Guidelines (MRP-227-A)*, December 2011, ADAMS Accession No. ML12017A193 (Transmittal letter from the EPRI-MRP) and ADAMS Accession Nos. ML12017A194, ML12017A196, ML12017A197, ML12017A191, ML12017A192, ML12017A195 and ML12017A199, (Final Report).

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

NRC Interim Staff Guidance LR-ISG-2011-05, *Ongoing Review Of Operating Experience*, March 16, 2012, (ADAMS Accession No. ML12044A215).

Nuclear Energy Institute (NEI) Report No. 03-08, Revision 2, *Guideline for the Management of Materials Issues*, ADAMS Accession No. ML101050334).

NRC 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*, December 16, 2011, ADAMS Accession No. ML11308A770.

(2) Revised version of GALL Report Chapter IV.B2

B2. REACTOR VESSEL INTERNALS (PWR) – WESTINGHOUSE

Systems, Structures, and Components

This section addresses the Westinghouse pressurized-water reactor (PWR) vessel internals, which consist of components in the upper internals assembly, the control rod guide tube 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.

Common miscellaneous material/environment combinations where aging effects are not expected to degrade the ability of the structure or component to perform its intended function for the period of extended operation are included in IV.E.

System Interfaces

The systems that interface with the reactor vessel internals include the reactor pressure vessel (IV.A2).

IV B2	REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM Reactor Vessel Internals (PWR) - Westinghouse						
Item	**Link**	**Structure and/or Component**	**Material**	**Environment**	**Aging Effect/ Mechanism**	**Aging Management Program (AMP)**	**Further Evaluation**
IV.B2.RP-300	IV.B2-33 (R-108)	Alignment and interfacing components: internals hold down spring	Stainless steel	Reactor coolant and neutron flux	Loss of preload due to thermal and irradiation enhanced stress relaxation; changes in dimensions due to void swelling or distortion; loss of material due to wear	Chapter XI.M16A, “PWR Vessel Internals”	No
IV.B2.RP-301	IV.B2-40 (R-112)	Alignment and interfacing components: upper core plate alignment pins	Stainless steel	Reactor coolant and neutron flux	Cracking due to stress corrosion cracking	Chapter XI.M16A, “PWR Vessel Internals” and Chapter XI.M2, “Water Chemistry”	No
IV.B2.RP-299	IV.B2-34 (R-115)	Alignment and interfacing components: upper core plate alignment pins	Stainless steel	Reactor coolant and neutron flux	Loss of material due to wear	Chapter XI.M16A, “PWR Vessel Internals”	No
IV.B2.RP-271	IV.B2-10 (R-125)	Baffle-to-former assembly: baffle-to-former bolts	Stainless steel	Reactor coolant and neutron flux	Cracking due to irradiation-assisted stress corrosion cracking or fatigue	Chapter XI.M16A, “PWR Vessel Internals” and Chapter XI.M2, “Water Chemistry” (for SCC mechanisms only)	No

IV B2	REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM Reactor Vessel Internals (PWR) - Westinghouse						
Item	**Link**	**Structure and/or Component**	**Material**	**Environment**	**Aging Effect/ Mechanism**	**Aging Management Program (AMP)**	**Further Evaluation**
IV.B2.RP-272	IV.B2-6 (R-128)	Baffle-to-former assembly: baffle-to-former bolts	Stainless steel	Reactor coolant and neutron flux	Loss of fracture toughness due to neutron irradiation embrittlement; changes in dimensions due to void swelling or distortion; loss of preload due to thermal and irradiation enhanced stress relaxation or creep	Chapter XI.M16A, “PWR Vessel Internals”	No
IV.B2.RP-270	IV.B2-1 (R-124)	Baffle-to-former assembly: baffle and former plates	Stainless steel	Reactor coolant and neutron flux	Changes in dimensions due to void swelling or distortion	Chapter XI.M16A, “PWR Vessel Internals”	No
IV.B2.RP-270a	IV.B2-1 (R-124)	Baffle-to-former assembly: baffle and former plates	Stainless steel	Reactor coolant and neutron flux	Cracking due to irradiation-assisted stress corrosion cracking	Chapter XI.M16A, “PWR Vessel Internals” and Chapter XI.M2, “Water Chemistry”	No
IV.B2.RP-275	IV.B2-6 (R-128)	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 stress corrosion cracking or fatigue	Chapter XI.M16A, “PWR Vessel Internals” and Chapter XI.M2, “Water Chemistry” (for SCC mechanisms only)	No

IV REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM
B2 Reactor Vessel Internals (PWR) - Westinghouse

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
IV.B2.RP-354		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	Chapter XI.M16A, “PWR Vessel Internals”	No
IV.B2.RP-273	IV.B2-10 (R-125)	Baffle-to-former assembly: barrel-to-former bolts	Stainless steel	Reactor coolant and neutron flux	Cracking due to irradiation-assisted stress corrosion cracking or fatigue	Chapter XI.M16A, “PWR Vessel Internals” and Chapter XI.M2, “Water Chemistry (for SCC mechanisms only)	No
IV.B2.RP-274	IV.B2-6 (R-128)	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	Chapter XI.M16A, “PWR Vessel Internals”	No

IV REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM
B2 Reactor Vessel Internals (PWR) - Westinghouse

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
IV.B2.RP-284	IV.B2-12 (R-143)	Bottom mounted instrument system: flux thimble tubes	Stainless steel (with or without chrome plating)	Reactor coolant and neutron flux	Loss of material due to wear	Chapter XI.M16A, “PWR Vessel Internals” or Chapter XI.M37,”Flux Thimble Tube Inspection”	No
IV.B2.RP-293	IV.B2-24 (R-138)	Bottom-mounted instrumentation system: bottom-mounted instrumentation (BMI) column bodies	Stainless steel	Reactor coolant and neutron flux	Cracking due to fatigue	Chapter XI.M16A, “PWR Vessel Internals”	No
IV.B2.RP-292		Bottom-mounted instrumentation system: bottom-mounted instrument (BMI) column bodies	Stainless steel	Reactor coolant and neutron flux	Loss of fracture toughness due to neutron irradiation embrittlement	Chapter XI.M16A, “PWR Vessel Internals”	No
IV.B2.RP-296		Control rod guide tube (CRGT) assemblies: CRGT guide plates (cards)	Stainless steel	Reactor coolant and neutron flux	Loss of material due to wear	Chapter XI.M16A, “PWR Vessel Internals”	No
IV.B2.RP-298	IV.B2-28 (R-118)	Control rod guide tube (CRGT) assemblies: CRGT lower flange welds	Stainless steel	Reactor coolant and neutron flux	Cracking due to stress corrosion cracking or fatigue	Chapter XI.M16A, “PWR Vessel Internals” and Chapter XI.M2, “Water Chemistry” (for SCC mechanisms only)	No
IV.B2.RP-297		Control rod guide tube (CRGT) assemblies: CRGT lower flange welds	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	Chapter XI.M16A, “PWR Vessel Internals”	No

IV REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM
B2 Reactor Vessel Internals (PWR) - Westinghouse

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
IV.B2.RP-355		Control rod guide tube (CRGT) assemblies: guide tube support pins (split pins)	Stainless steel; nickel alloy	Reactor coolant and neutron flux	Cracking due to stress corrosion cracking or fatigue	Chapter XI.M16A, “PWR Vessel Internals” and Chapter XI.M2, “Water Chemistry” (for SCC mechanisms only)	No
IV.B2.RP-356		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	Chapter XI.M16A, “PWR Vessel Internals”	No
IV.B2.RP-387		Core barrel assembly: upper core barrel and lower core barrel circumferential (girth) welds	Stainless steel	Reactor coolant and neutron flux	Cracking due to stress corrosion cracking or irradiation-assisted stress corrosion cracking or fatigue	Chapter XI.M16A, “PWR Vessel Internals” and Chapter XI.M2, “Water Chemistry” (for SCC mechanisms only)	No
IV.B2.RP-387a		Core barrel assembly: upper core barrel and lower core barrel vertical (axial) welds	Stainless steel	Reactor coolant and neutron flux	Cracking due to stress corrosion cracking or irradiation-assisted stress corrosion cracking or fatigue	Chapter XI.M16A, “PWR Vessel Internals” and Chapter XI.M2, “Water Chemistry” (for SCC mechanisms only)	No
IV.B2.RP-388		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	Chapter XI.M16A, “PWR Vessel Internals”	No
IV.B2.RP-388a		Core barrel assembly: upper core barrel and lower core barrel vertical (axial) welds	Stainless steel	Reactor coolant and neutron flux	Loss of fracture toughness due to neutron irradiation embrittlement	Chapter XI.M16A, “PWR Vessel Internals”	No

IV REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM
B2 Reactor Vessel Internals (PWR) - Westinghouse

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
IV.B2.RP-345		Core barrel assembly: core barrel flange	Stainless steel	Reactor coolant and neutron flux	Loss of material due to wear	Chapter XI.M16A, “PWR Vessel Internals”	No
IV.B2.RP-278	IV.B2-8 (R-120)	Core barrel assembly: core barrel outlet nozzle welds	Stainless steel	Reactor coolant and neutron flux	Cracking due to stress corrosion cracking or fatigue	Chapter XI.M16A, “PWR Vessel Internals” and Chapter XI.M2, “Water Chemistry” (for SCC mechanisms only)	No
IV.B2.RP-278a		Core barrel assembly: core barrel outlet nozzle welds	Stainless steel	Reactor coolant and neutron flux	Loss of fracture toughness due to neutron irradiation embrittlement	Chapter XI.M16A, “PWR Vessel Internals”	No
IV.B2.RP-280	IV.B2-8 (R-120)	Core barrel assembly: lower core barrel flange weld	Stainless steel	Reactor coolant and neutron flux	Cracking due to stress corrosion cracking or fatigue	Chapter XI.M16A, “PWR Vessel Internals” and Chapter XI.M2, “Water Chemistry” (for SCC mechanisms only)	No
IV.B2.RP-276	IV.B2-8 (R-120)	Core barrel assembly: upper core barrel flange weld	Stainless steel	Reactor coolant and neutron flux	Cracking due to stress corrosion cracking and irradiation-assisted stress corrosion cracking	Chapter XI.M16A, “PWR Vessel Internals” and Chapter XI.M2, “Water Chemistry”	No
IV.B2.RP-285	IV.B2-14 (R-137)	Lower internals assembly: clevis insert bolts or screws	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	Chapter XI.M16A, “PWR Vessel Internals”	No

IV B2	REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM Reactor Vessel Internals (PWR) - Westinghouse						
Item	**Link**	**Structure and/or Component**	**Material**	**Environment**	**Aging Effect/ Mechanism**	**Aging Management Program (AMP)**	**Further Evaluation**
IV.B2.RP-399		Lower internals assembly: clevis insert bolts or screws	Stainless steel; nickel alloy	Reactor coolant and neutron flux	Cracking due to primary water stress corrosion cracking, irradiation-assisted stress corrosion cracking or fatigue	Chapter XI.M16A, “PWR Vessel Internals” and Chapter XI.M2, “Water Chemistry” (for SCC mechanisms only)	No
IV.B2.RP-289	IV.B2-20 (R-130)	Lower internals assembly: lower core plate and extra-long (XL) lower core plate	Stainless steel	Reactor coolant and neutron flux	Cracking due to irradiation-assisted stress corrosion cracking or fatigue	Chapter XI.M16A, “PWR Vessel Internals” and Chapter XI.M2, “Water Chemistry” (for SCC mechanisms only)	No
IV.B2.RP-288	IV.B2-18 (R-132)	Lower internals assembly: lower core plate and 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	Chapter XI.M16A, “PWR Vessel Internals”	No
IV.B2.RP-291	IV.B2-24 (R-138)	Lower support assembly: lower support column bodies (cast)	Cast austenitic stainless steel	Reactor coolant and neutron flux	Cracking due to irradiation-assisted stress corrosion cracking	Chapter XI.M16A, “PWR Vessel Internals” and Chapter XI.M2, “Water Chemistry”	No
IV.B2.RP-290	IV.B2-21 (R-140)	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	Chapter XI.M16A, “PWR Vessel Internals”	No

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B2 Reactor Vessel Internals (PWR) - Westinghouse

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
IV.B2.RP-291a		Lower support assembly: lower support forging or casting	Stainless steel	Reactor coolant and neutron flux	Cracking due to stress corrosion cracking or fatigue	Chapter XI.M16A, “PWR Vessel Internals” and Chapter XI.M2, “Water Chemistry” (for SCC mechanisms only)	No
IV.B2.RP-290a		Lower support assembly: lower support forging or casting	Stainless steel	Reactor coolant and neutron flux	Loss of fracture toughness due to neutron irradiation embrittlement (and thermal aging embrittlement for CASS, PH SS, and martensitic SS)	Chapter XI.M16A, “PWR Vessel Internals”	No
IV.B2.RP-294	IV.B2-24 (R-138)	Lower support assembly: lower support column bodies (non-cast)	Stainless steel	Reactor coolant and neutron flux	Cracking due to irradiation-assisted stress corrosion cracking	Chapter XI.M16A, “PWR Vessel Internals” and Chapter XI.M2, “Water Chemistry”	No
IV.B2.RP-295		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	Chapter XI.M16A, “PWR Vessel Internals”	No
IV.B2.RP-286	IV.B2-16 (R-133)	Lower support assembly: lower support column bolts	Stainless steel	Reactor coolant and neutron flux	Cracking due to irradiation-assisted stress corrosion cracking or fatigue	Chapter XI.M16A, “PWR Vessel Internals” and Chapter XI.M2, “Water Chemistry” (for SCC mechanisms only)	No

IV B2	REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM Reactor Vessel Internals (PWR) - Westinghouse						
Item	**Link**	**Structure and/or Component**	**Material**	**Environment**	**Aging Effect/ Mechanism**	**Aging Management Program (AMP)**	**Further Evaluation**
IV.B2.RP-287	IV.B2-17 (R-135)	Lower support assembly: lower support column 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	Chapter XI.M16A, “PWR Vessel Internals”	No
IV.B2.RP-303	IV.B2-31 (R-53)	Reactor vessel internal components	Stainless steel; nickel alloy	Reactor coolant and neutron flux	Cumulative fatigue damage due to fatigue	Fatigue is a time-limited aging analysis (TLAA) to be evaluated for the period of extended operation. See the SRP, Section 4.3 “Metal Fatigue,” for acceptable methods for meeting the requirements of 10 CFR 54.21(c)(1).	Yes – TLAA
IV.B2.RP-24	IV.B2-32 (RP-24)	Reactor vessel internal components	Stainless steel; nickel alloy	Reactor coolant and neutron flux	Loss of material due to pitting and crevice corrosion	Chapter XI.M2, “Water Chemistry”	No

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B2 Reactor Vessel Internals (PWR) - Westinghouse

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
IV.B2.RP-382	IV.B2-26 (R-142)	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	Reactor coolant and neutron flux	Cracking due to fatigue, stress corrosion cracking, or irradiation-assisted stress corrosion cracking; loss of material due to wear	Chapter XI.M1, "ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD" or Chapter XI.M16A, "PWR Vessel Internals," by invoking applicable 10 CFR 50.55a and ASME Section XI inservice inspection requirements	No
IV.B2.RP-302		Thermal shield assembly: thermal shield flexures	Stainless steel	Reactor coolant and neutron flux	Cracking due to fatigue	Chapter XI.M16A, "PWR Vessel Internals"	No
IV.B2.RP-302a		Thermal shield assembly: thermal shield flexures	Stainless steel	Reactor coolant and neutron flux	Loss of material due to wear	Chapter XI.M16A, "PWR Vessel Internals"	No
IV.B2.RP-265		Reactor internal "No Additional Measures" components	Stainless steel; nickel alloy	Reactor coolant and neutron flux	No additional aging management for reactor internal "No Additional Measures" components unless required by ASME Section XI, Examination Category B-N-3 or relevant operating experience exists	Chapter XI.M16A, "PWR Vessel Internals"	No
IV.B2.RP-291b		Upper Internals Assembly; upper core plate	Stainless steel	Reactor coolant and neutron flux	Cracking due to fatigue	Chapter XI.M16A, "PWR Vessel Internals"	No

IV B2	REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM Reactor Vessel Internals (PWR) - Westinghouse						
Item	**Link**	**Structure and/or Component**	**Material**	**Environment**	**Aging Effect/ Mechanism**	**Aging Management Program (AMP)**	**Further Evaluation**
IV.B2.RP-290b		Upper Internals Assembly; upper core plate	Stainless steel	Reactor coolant and neutron flux	Loss of material due to wear	Chapter XI.M16A, “PWR Vessel Internals”	No
IV.B2.RP-346		Upper Internals Assembly: upper support ring or skirt	Stainless steel	Reactor coolant and neutron flux	Cracking due to stress corrosion cracking or fatigue	Chapter XI.M16A, “PWR Vessel Internals” and Chapter XI.M2, “Water Chemistry” (for SCC mechanisms only)	No

(3) Revised version of GALL Report Chapter IV.B3

B3. REACTOR VESSEL INTERNALS (PWR) - COMBUSTION ENGINEERING

Systems, Structures, and Components

This section 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 encore instrumentation (ICI) components.

Common miscellaneous material/environment combinations where aging effects are not expected to degrade the ability of the structure or component to perform its intended function for the period of extended operation are included in IV.E.

System Interfaces

The systems that interface with the reactor vessel internals include the reactor pressure vessel (IV.A2).

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B3 Reactor Vessel Internals (PWR) - Combustion Engineering

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
IV.B3.RP-312	IV.B3-2 (R-149)	Control Element Assembly (CEA): instrument guide tubes in peripheral CEA assemblies	Stainless steel	Reactor coolant and neutron flux	Cracking due to stress corrosion cracking or fatigue	Chapter XI.M16A, “PWR Vessel Internals” and Chapter XI.M2, “Water Chemistry” (for SCC mechanisms only)	No
IV.B3.RP-313		Control Element Assembly (CEA): remaining instrument guide tubes in CEA assemblies	Stainless steel	Reactor coolant and neutron flux	Cracking due to stress corrosion cracking or fatigue	Chapter XI.M16A, “PWR Vessel Internals” and Chapter XI.M2, “Water Chemistry” (for SCC mechanisms only)	No
IV.B3.RP-320	IV.B3-9 (R-162)	Core shroud assemblies (all plants): guide lugs; guide lug inserts and bolts	Stainless steel	Reactor coolant and neutron flux	Cracking due to fatigue	Chapter XI.M16A, “PWR Vessel Internals”	No
IV.B3.RP-319	IV.B3-9 (R-162)	Core shroud assemblies (all plants): guide lugs; 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	Chapter XI.M16A, "PWR Vessel Internals”	No

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B3 Reactor Vessel Internals (PWR) - Combustion Engineering

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
IV.B3.RP-358		Core shroud assemblies (for bolted core shroud assemblies): assembly components, including shroud plates and former plates	Stainless steel	Reactor coolant and neutron flux	Cracking due to irradiation-assisted stress corrosion cracking	Chapter XI.M16A, “PWR Vessel Internals” and Chapter XI.M2, “Water Chemistry”	No
IV.B3.RP-318	IV.B3-8 (R-163)	Core shroud assemblies (for bolted core shroud assemblies): assembly components, including shroud plates and former plates	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	Chapter XI.M16A, "PWR Vessel Internals"	No
IV.B3.RP-316	IV.B3-9 (R-162)	Core shroud assemblies (for bolted core shroud assemblies): barrel-shroud bolts	Stainless steel	Reactor coolant and neutron flux	Cracking due to irradiation-assisted stress corrosion cracking or fatigue	Chapter XI.M16A, “PWR Vessel Internals” and Chapter XI.M2, “Water Chemistry” (for SCC mechanisms only)	No

IV B3	REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM Reactor Vessel Internals (PWR) - Combustion Engineering						
Item	**Link**	**Structure and/or Component**	**Material**	**Environment**	**Aging Effect/ Mechanism**	**Aging Management Program (AMP)**	**Further Evaluation**
IV.B3.RP-317	IV.B3-7 (R-165)	Core shroud assemblies (for bolted core shroud assemblies): barrel-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	Chapter XI.M16A, "PWR Vessel Internals"	No
IV.B3.RP-314	IV.B3-9 (R-162)	Core shroud assemblies (for bolted core shroud assemblies): core shroud bolts	Stainless steel	Reactor coolant and neutron flux	Cracking due to irradiation-assisted stress corrosion cracking or fatigue	Chapter XI.M16A, “PWR Vessel Internals” and Chapter XI.M2, “Water Chemistry” (for SCC mechanisms only)	No
IV.B3.RP-315	IV.B3-7 (R-165)	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 dimensions due to void swelling or distortion	Chapter XI.M16A, “PWR Vessel Internals”	No

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Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
IV.B3.RP-359		Core shroud assembly (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	Chapter XI.M16A, “PWR Vessel Internals”	No
IV.B3.RP-322		Core shroud assembly (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 stress corrosion cracking	Chapter XI.M16A, “PWR Vessel Internals” and Chapter XI.M2, “Water Chemistry”	No
IV.B3.RP-326		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	Changes in dimensions due to void swelling or distortion; loss of fracture toughness due to neutron irradiation embrittlement	Chapter XI.M16A, "PWR Vessel Internals"	No

IV B3	REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM Reactor Vessel Internals (PWR) - Combustion Engineering						
Item	**Link**	**Structure and/or Component**	**Material**	**Environment**	**Aging Effect/ Mechanism**	**Aging Management Program (AMP)**	**Further Evaluation**
IV.B3.RP-326a		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 stress corrosion cracking or fatigue	Chapter XI.M16A, “PWR Vessel Internals” and Chapter XI.M2, “Water Chemistry” (for SCC mechanisms only)	No
IV.B3.RP-323		Core shroud assembly (designs assembled in two vertical sections): remaining axial welds	Stainless steel	Reactor coolant and neutron flux	Cracking due to irradiation-assisted stress corrosion cracking	Chapter XI.M16A, “PWR Vessel Internals” and Chapter XI.M2, “Water Chemistry”	No
IV.B3.RP-359a		Core shroud assembly (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	Chapter XI.M16A, “PWR Vessel Internals”	No

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B3 Reactor Vessel Internals (PWR) - Combustion Engineering

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
IV.B3.RP-324		Core shroud assembly (designs assembled with full-height shroud plates): shroud plate axial weld seams at the core shroud re-entrant corners	Stainless steel	Reactor coolant and neutron flux	Cracking due to irradiation-assisted stress corrosion cracking	Chapter XI.M16A, “PWR Vessel Internals” and Chapter XI.M2, “Water Chemistry”	No
IV.B3.RP-360		Core shroud assembly (designs assembled with full-height shroud plates): shroud plate axial weld seams at the core shroud re-entrant corners	Stainless steel	Reactor coolant and neutron flux	Loss of fracture toughness due to neutron irradiation embrittlement	Chapter XI.M16A, "PWR Vessel Internals"	No
IV.B3.RP-325		Core shroud assembly (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 stress corrosion cracking	Chapter XI.M16A, “PWR Vessel Internals” and Chapter XI.M2, “Water Chemistry””	No

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

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
IV.B3.RP-361		Core shroud assembly (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	Chapter XI.M16A, "PWR Vessel Internals"	No
IV.B3.RP-362		Core support barrel assembly: lower cylinder circumferential (girth) welds	Stainless steel	Reactor coolant and neutron flux	Loss of fracture toughness due to neutron irradiation embrittlement	Chapter XI.M16A, "PWR Vessel Internals"	No
IV.B3.RP-362a		Core support barrel assembly: lower cylinder circumferential (girth) welds	Stainless steel	Reactor coolant and neutron flux	Cracking due to stress corrosion cracking or irradiation-assisted stress corrosion cracking	Chapter XI.M16A, “PWR Vessel Internals” and Chapter XI.M2, “Water Chemistry”	No
IV.B3.RP-362b		Core support barrel assembly: lower cylinder vertical (axial) welds	Stainless steel	Reactor coolant and neutron flux	Loss of fracture toughness due to neutron irradiation embrittlement	Chapter XI.M16A, "PWR Vessel Internals"	No

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B3 Reactor Vessel Internals (PWR) - Combustion Engineering

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
IV.B3.RP-362c		Core support barrel assembly: lower cylinder vertical (axial) welds	Stainless steel	Reactor coolant and neutron flux	Cracking due to stress corrosion cracking or irradiation-assisted stress corrosion cracking	Chapter XI.M16A, “PWR Vessel Internals” and Chapter XI.M2, “Water Chemistry”	No
IV.B3.RP-329	IV.B3-15 (R-155)	Core support barrel assembly: upper cylinder (base metal and welds) and upper core barrel flange (flange base metal)	Stainless steel	Reactor coolant and neutron flux	Cracking due to stress corrosion cracking	Chapter XI.M16A, “PWR Vessel Internals” and Chapter XI.M2, “Water Chemistry”	No
IV.B3.RP-333		Core support barrel assembly: lower flange	Stainless steel	Reactor coolant and neutron flux	Cracking due to stress corrosion cracking or fatigue	Chapter XI.M16A, “PWR Vessel Internals” and Chapter XI.M2, “Water Chemistry” (for SCC mechanisms only)	No
IV.B3.RP-328	IV.B3-15 (R-155)	Core support barrel assembly: lower core barrel flange weld	Stainless steel	Reactor coolant and neutron flux	Cracking due to stress corrosion cracking or fatigue	Chapter XI.M16A, “PWR Vessel Internals” and Chapter XI.M2, “Water Chemistry” (for SCC mechanisms only)	No
IV.B3.RP-332	IV.B3-17 (R-156)	Core support barrel assembly: upper core barrel flange	Stainless steel	Reactor coolant and neutron flux	Loss of material due to wear	Chapter XI.M16A, “PWR Vessel Internals”	No
IV.B3.RP-327	IV.B3-15 (R-155)	Core support barrel assembly: upper core barrel flange weld	Stainless steel	Reactor coolant and neutron flux	Cracking due to stress corrosion cracking	Chapter XI.M16A, “PWR Vessel Internals” and Chapter XI.M2, “Water Chemistry”	No

IV REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM B3 Reactor Vessel Internals (PWR) - Combustion Engineering							
Item	**Link**	**Structure and/or Component**	**Material**	**Environment**	**Aging Effect/ Mechanism**	**Aging Management Program (AMP)**	**Further Evaluation**
IV.B3.RP-357		Incore instruments (ICI): ICI thimble tubes - lower	Zircaloy-4	Reactor coolant and neutron flux	Loss of material due to wear	Chapter XI.M16A, “PWR Vessel Internals”	No
IV.B3.RP-336	IV.B3-22 (R-170)	Lower support structure (designs assembled in two vertical sections): 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	Chapter XI.M16A, “PWR Vessel Internals”	No
IV.B3.RP-334	IV.B3-23 (R-167)	Lower support structure (designs assembled in two vertical sections or with full-height shroud plates): fuel alignment pins	Stainless steel	Reactor coolant and neutron flux	Cracking due to stress corrosion cracking, irradiation-assisted stress corrosion cracking, or fatigue	Chapter XI.M16A, “PWR Vessel Internals” and Chapter XI.M2, “Water Chemistry” (for SCC mechanisms only)	No

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

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
IV.B3.RP-334a	IV.B3-22 (R-170)	Lower support structure (designs assembled in two vertical sections or 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	Chapter XI.M16A, “PWR Vessel Internals”	No
IV.B3.RP-364		Lower support structure (all plants): core support column welds	Stainless steel (including CASS)	Reactor coolant and neutron flux	Loss of fracture toughness due to neutron irradiation embrittlement and for column welds made from CASS, thermal embrittlement	Chapter XI.M16A, "PWR Vessel Internals"	No
IV.B3.RP-363		Lower support structure (all plants): core support column welds	Stainless steel	Reactor coolant and neutron flux	Cracking due to stress corrosion cracking, irradiation-assisted stress corrosion cracking, or fatigue	Chapter XI.M16A, “PWR Vessel Internals” and Chapter XI.M2, “Water Chemistry” (for SCC mechanisms only)	No
IV.B3.RP-330	IV.B3-23 (R-167)	Lower support structure: core support column bolts	Stainless steel	Reactor coolant and neutron flux	Cracking due to irradiation-assisted stress corrosion cracking or fatigue	Chapter XI.M16A, “PWR Vessel Internals” and Chapter XI.M2, “Water Chemistry” (for SCC mechanisms only)	No

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

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
IV.B3.RP-331		Lower support structure: core support column bolts	Stainless steel	Reactor coolant and neutron flux	Loss of fracture toughness due to neutron irradiation embrittlement	Chapter XI.M16A, “PWR Vessel Internals”	No
IV.B3.RP-335	IV.B3-23 (R-167)	Lower support structure (designs except those assembled with full-height shroud plates): lower core support beams	Stainless steel	Reactor coolant and neutron flux	Cracking due to stress corrosion cracking or fatigue	Chapter XI.M16A, “PWR Vessel Internals” and Chapter XI.M2, “Water Chemistry” (for SCC mechanisms only)	No
IV.B3.RP-365		Lower support structure (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	Chapter XI.M16A, "PWR Vessel Internals"	No
IV.B3.RP-343		Lower support structure (designs with a core support plate): core support plate	Stainless steel	Reactor coolant and neutron flux	Cracking due to fatigue	Chapter XI.M16A, “PWR Vessel Internals”	No

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

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
IV.B3.RP-342		Lower support structure (designs with core shrouds assembled with full height shroud plates): deep beams	Stainless steel	Reactor coolant and neutron flux	Cracking due to stress corrosion cracking, irradiation-assisted stress corrosion cracking, or fatigue	Chapter XI.M16A, “PWR Vessel Internals” and Chapter XI.M2, “Water Chemistry” (for SCC mechanisms only)	No
IV.B3.RP-366		Lower support structure (designs with core shrouds assembled with full height shroud plates): deep beams	Stainless steel	Reactor coolant and neutron flux	Loss of fracture toughness due to neutron irradiation embrittlement	Chapter XI.M16A, “PWR Vessel Internals”	No
IV.B3.RP-339	IV.B3-24 (R-53)	Reactor vessel internal components	Stainless steel; nickel alloy	Reactor coolant and neutron flux	Cumulative fatigue damage due to fatigue	Fatigue is a time-limited aging analysis (TLAA) to be evaluated for the period of extended operation. See the SRP, Section 4.3 “Metal Fatigue,” for acceptable methods for meeting the requirements of 10 CFR 54.21(c)(1).	Yes, TLAA
IV.B3.RP-306		Reactor internal “No Additional Measures” components	Stainless steel; nickel alloy	Reactor coolant and neutron flux	No additional aging management for reactor internal “No Additional Measures” components unless required by ASME Section XI, Examination Category B-N-3 or relevant operating experience exists	Chapter XI.M16A, “PWR Vessel Internals”	No

IV B3	REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM Reactor Vessel Internals (PWR) - Combustion Engineering						
Item	**Link**	**Structure and/or Component**	**Material**	**Environment**	**Aging Effect/ Mechanism**	**Aging Management Program (AMP)**	**Further Evaluation**
IV.B3.RP-24	IV.B3-25 (RP-24)	Reactor vessel internal components	Stainless steel; nickel alloy	Reactor coolant and neutron flux	Loss of material due to pitting and crevice corrosion	Chapter XI.M2, “Water Chemistry”	No
IV.B3.RP-382	IV.B3-22 (R-170)	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	Reactor coolant and neutron flux	Cracking due to fatigue, stress corrosion cracking, or irradiation-assisted stress corrosion cracking; loss of material due to wear	Chapter XI.M1, “ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD” or Chapter XI.M16A, “PWR Vessel Internals,” by invoking applicable 10 CFR 50.55a and ASME Section XI inservice inspection requirements	No
IV.B3.RP-338		Upper internals assembly (designs with core shrouds assembled with full height shroud plates): fuel alignment plate	Stainless steel	Reactor coolant and neutron flux	Cracking due to fatigue	Chapter XI.M16A, “PWR Vessel Internals”	No
IV.B3.RP-400		Core Support Barrel Assembly: thermal shield positioning pins	Stainless steel	Reactor coolant and neutron flux	Cracking due to stress corrosion cracking, irradiation-assisted stress corrosion cracking or fatigue; loss of material due to wear	Chapter XI.M16A, “PWR Vessel Internals” and Chapter XI.M2, “Water Chemistry” (for SCC mechanisms only)	No

(4) Revised version of GALL Report Chapter IV.B4

B4. REACTOR VESSEL INTERNALS (PWR) - BABCOCK AND WILCOX

Systems, Structures, and Components

This section addresses the Babcock and 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, incore monitoring instrumentation (IMI) guide tube assembly, and the flow distributor assembly.

Common miscellaneous material/environment combinations where aging effects are not expected to degrade the ability of the structure or component to perform its intended function for the period of extended operation are included in IV.E.

System Interfaces

The systems that interface with the reactor vessel internals include the reactor pressure vessel (IV.A2).

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

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
IV.B4.RP-242	IV.B4-4 (R-183)	Control rod guide tube (CRGT) assembly: CRGT spacer castings	Cast austenitic stainless steel	Reactor coolant and neutron flux	Loss of fracture toughness due to thermal aging embrittlement	Chapter XI.M16A, “PWR Vessel Internals”	No
IV.B4.RP-242a		Control rod guide tube (CRGT) assembly: CRGT spacer castings	Stainless steel (including CASS)	Reactor coolant and neutron flux	Cracking due to stress corrosion cracking or fatigue	Chapter XI.M16A, “PWR Vessel Internals” and Chapter XI.M2, “Water Chemistry” (for SCC mechanisms only)	No
IV.B4.RP-245	IV.B4-13 (R-194)	Core barrel assembly (applicable to Crystal River Unit 3 or Davis Besse only): surveillance specimen holder tube (SSHT) studs/nuts or bolts	Nickel alloy	Reactor coolant and neutron flux	Cracking due to stress corrosion cracking	Chapter XI.M16A, “PWR Vessel Internals” and Chapter XI.M2, “Water Chemistry”	No
IV.B4.RP-245a		Core barrel assembly (applicable to Crystal River Unit 3 or Davis Besse only): surveillance specimen holder tube (SSHT) stud or bolt locking devices	Nickel alloy	Reactor coolant and neutron flux	Cracking due to fatigue	Chapter XI.M16A, “PWR Vessel Internals”	No

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B4 Reactor Vessel Internals (PWR) – Babcock & Wilcox

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
IV.B4.RP-245b		Core barrel assembly (applicable to CR-3 or DB only): surveillance specimen holder tube (SSHT) stud or bolt locking devices	Nickel alloy	Reactor coolant and neutron flux	Loss of material due to wear; changes in dimensions due to void swelling or distortion	Chapter XI.M16A, "PWR Vessel Internals"	No
IV.B4.RP-247	IV.B4-13 (R-194)	Core barrel assembly: lower core barrel (LCB) bolts	Stainless steel; nickel alloy	Reactor coolant and neutron flux	Cracking due to stress corrosion cracking	Chapter XI.M16A, "PWR Vessel Internals" and Chapter XI.M2, "Water Chemistry"	No
IV.B4.RP-247a		Core barrel assembly: lower core barrel (LCB) bolt locking devices	Stainless steel; nickel alloy	Reactor coolant and neutron flux	Cracking due to fatigue	Chapter XI.M16A, "PWR Vessel Internals"	No
IV.B4.RP-247b		Core barrel assembly: lower core barrel (LCB) bolt locking devices	Stainless steel; nickel alloy	Reactor coolant and neutron flux	Loss of material due to wear; changes in dimensions due to void swelling or distortion	Chapter XI.M16A, "PWR Vessel Internals"	No
IV.B4.RP-249	IV.B4-12 (R-196)	Core barrel assembly: baffle plates	Stainless steel	Reactor coolant and neutron flux	Loss of fracture toughness due to neutron irradiation embrittlement	Chapter XI.M16A, "PWR Vessel Internals"	No
IV.B4.RP-249a		Core barrel assembly: baffle plates	Stainless steel	Reactor coolant and neutron flux	Cracking due to irradiation-assisted stress corrosion cracking	Chapter XI.M16A, "PWR Vessel Internals" and Chapter XI.M2, "Water Chemistry"	No

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B4 Reactor Vessel Internals (PWR) – Babcock & Wilcox

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
IV.B4.RP-241	IV.B4-7 (R-125)	Core barrel assembly: baffle-to-former bolts and screws	Stainless steel	Reactor coolant and neutron flux	Cracking due to stress corrosion cracking, irradiation-assisted stress corrosion cracking, fatigue, and overload	Chapter XI.M16A, “PWR Vessel Internals” and Chapter XI.M2, “Water Chemistry” (for SCC mechanisms only)	No
IV.B4.RP-241a		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 stress corrosion cracking, irradiation-assisted stress corrosion cracking, fatigue, and overload	Chapter XI.M16A, “PWR Vessel Internals” and Chapter XI.M2, “Water Chemistry” (for SCC mechanisms only)	No
IV.B4.RP-240	IV.B4-1 (R-128)	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 or creep; loss of material due to wear	Chapter XI.M16A, “PWR Vessel Internals”	No

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

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
IV.B4.RP-240a		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	Chapter XI.M16A, “PWR Vessel Internals”	No
IV.B4.RP-250	IV.B4-12 (R-196)	Core barrel assembly: core barrel cylinder (including vertical and circumferential seam welds); former plates	Stainless steel	Reactor coolant and neutron flux	Loss of fracture toughness due to neutron irradiation embrittlement	Chapter XI.M16A, “PWR Vessel Internals”	No
IV.B4.RP-250a		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 stress corrosion cracking or fatigue	Chapter XI.M16A, “PWR Vessel Internals” and Chapter XI.M2, “Water Chemistry” (for SCC mechanisms only)	No
IV.B4.RP-375		Core barrel assembly: internal baffle-to-baffle bolts	Stainless steel	Reactor coolant and neutron flux	Cracking due to irradiation-assisted stress corrosion cracking, fatigue, or overload	Chapter XI.M16A, “PWR Vessel Internals” and Chapter XI.M2, “Water Chemistry” (for SCC mechanisms only)	No

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B4 Reactor Vessel Internals (PWR) – Babcock & Wilcox

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
IV.B4.RP-375a		Core barrel assembly: internal baffle-to-baffle 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	Chapter XI.M16A, "PWR Vessel Internals"	No
IV.B4.RP-244	IV.B4-7 (R-125)	Core barrel assembly; external baffle-to-baffle bolts and core barrel-to-former bolts;	Stainless steel	Reactor coolant and neutron flux	Cracking due to irradiation-assisted stress corrosion cracking, fatigue, and overload	Chapter XI.M16A, "PWR Vessel Internals" and Chapter XI.M2, "Water Chemistry" (for SCC mechanisms only)	No
IV.B4.RP-244a		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 stress corrosion cracking, or fatigue	Chapter XI.M16A, "PWR Vessel Internals" and Chapter XI.M2, "Water Chemistry" (for SCC mechanisms only)	No

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B4 Reactor Vessel Internals (PWR) – Babcock & Wilcox

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
IV.B4.RP-243	IV.B4-1 (R-128)	Core barrel assembly: 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 preload due to thermal and irradiation enhanced stress relaxation or creep; loss of material due to wear	Chapter XI.M16A, "PWR Vessel Internals"	No
IV.B4.RP-243a		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	Loss of fracture toughness due to neutron irradiation embrittlement; loss of material due to wear	Chapter XI.M16A, "PWR Vessel Internals"	No
IV.B4.RP-248	IV.B4-12 (R-196)	Core support shield (CSS) assembly: upper core barrel (UCB) bolts	Stainless steel; nickel alloy	Reactor coolant and neutron flux	Cracking due to stress corrosion cracking	Chapter XI.M16A, “PWR Vessel Internals” and Chapter XI.M2, “Water Chemistry”	No
IV.B4.RP-248a		Core support shield (CSS) assembly: upper core barrel (UCB) bolt locking devices	Stainless steel; nickel alloy	Reactor coolant and neutron flux	Cracking due to fatigue	Chapter XI.M16A, “PWR Vessel Internals”	No

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B4 Reactor Vessel Internals (PWR) – Babcock & Wilcox

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
IV.B4.RP-248b		Core support shield (CSS) assembly: upper core barrel (UCB) bolt locking devices	Stainless steel; nickel alloy	Reactor coolant and neutron flux	Loss of material due to wear; changes in dimensions due to void swelling or distortion	Chapter XI.M16A, "PWR Vessel Internals"	No
IV.B4.RP-252	IV.B4-16 (R-188)	Core support shield (CSS) assembly: CSS vent valve top and bottom retaining rings (valve body components)	Stainless steel, including CASS and PH steels	Reactor coolant and neutron flux	Loss of fracture toughness due to thermal aging embrittlement	Chapter XI.M16A, "PWR Vessel Internals"	No
IV.B4.RP-252a	IV.B4-16 (R-188)	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 stress corrosion cracking or fatigue	Chapter XI.M16A, "PWR Vessel Internals" and Chapter XI.M2, "Water Chemistry" (for SCC mechanisms only)	No
IV.B4.RP-251	IV.B4-15 (R-190)	Core support shield (CSS) assembly: CSS top flange	Stainless steel	Reactor coolant and neutron flux	Loss of material due to wear; loss of preload (wear)	Chapter XI.M16A, "PWR Vessel Internals"	No
IV.B4.RP-251a	IV.B4-15 (R-190)	Plenum cover assembly: plenum cover weldment rib pads and plenum cover support flange	Stainless steel	Reactor coolant and neutron flux	Loss of material due to wear; loss of preload (wear)	Chapter XI.M16A, "PWR Vessel Internals"	No

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B4 Reactor Vessel Internals (PWR) – Babcock & Wilcox

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
IV.B4.RP-256	IV.B4-25 (R-210)	Flow distributor assembly: flow distributor bolts	Stainless steel; nickel alloy	Reactor coolant and neutron flux	Cracking due to stress corrosion cracking	Chapter XI.M16A, “PWR Vessel Internals” and Chapter XI.M2, “Water Chemistry”	No
IV.B4.RP-256a		Flow distributor assembly: flow distributor bolt locking devices	Stainless steel; nickel alloy	Reactor coolant and neutron flux	Cracking due to fatigue	Chapter XI.M16A, “PWR Vessel Internals”	No
IV.B4.RP-256b		Flow distributor assembly: flow distributor 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	Chapter XI.M16A, “PWR Vessel Internals”	No
IV.B4.RP-259	IV.B4-31 (R-205)	Incore Monitoring Instrument (IMI) guide tube assembly: IMI guide tube spider-to-lower grid rib section welds	Stainless steel	Reactor coolant and neutron flux	Loss of fracture toughness due to neutron irradiation embrittlement	Chapter XI.M16A, “PWR Vessel Internals”	No
IV.B4.RP-259a		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 stress corrosion cracking, irradiation-assisted stress corrosion cracking or fatigue	Chapter XI.M16A, “PWR Vessel Internals” and Chapter XI.M2, “Water Chemistry”	No.

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B4 Reactor Vessel Internals (PWR) – Babcock & Wilcox

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
IV.B4.RP-258	IV.B4-4 (R-183)	Incore Monitoring Instrument (IMI) guide tube assembly: IMI guide tube spiders (castings)	Cast austenitic stainless steel	Reactor coolant and neutron flux	Loss of fracture toughness due to thermal aging and neutron irradiation embrittlement	Chapter XI.M16A, “PWR Vessel Internals”	No
IV.B4.RP-258a		Incore Monitoring Instrumentation (IMI) guide tube assembly: IMI guide tube spiders	Stainless steel	Reactor coolant and neutron flux	Cracking due to stress corrosion cracking, irradiation-assisted stress corrosion cracking or fatigue	Chapter XI.M16A, “PWR Vessel Internals” and Chapter XI.M2, “Water Chemistry”	No
IV.B4.RP-254	IV.B4-25 (R-210)	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 stress corrosion cracking	Chapter XI.M16A, “PWR Vessel Internals” and Chapter XI.M2, “Water Chemistry”	No
IV.B4.RP-254a		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	Chapter XI.M16A, “PWR Vessel Internals”	No
IV.B4.RP-254b		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	Chapter XI.M16A, “PWR Vessel Internals”	No

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

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
IV.B4.RP-246	IV.B4-12 (R-196)	Lower grid assembly: upper thermal shield (UTS) bolts and lower thermal shield (LTS) bolts	Stainless steel; nickel alloy	Reactor coolant and neutron flux	Cracking due to stress corrosion cracking	Chapter XI.M16A, “PWR Vessel Internals” and Chapter XI.M2, “Water Chemistry”	No
IV.B4.RP-246a		Lower grid assembly: upper thermal shield (UTS) bolt locking devices and lower thermal shield (LTS) bolt locking devices	Stainless steel; nickel alloy	Reactor coolant and neutron flux	Cracking due to fatigue	Chapter XI.M16A, “PWR Vessel Internals”	No
IV.B4.RP-246b		Lower grid assembly: upper thermal shield (UTS) bolt locking devices and lower thermal shield (LTS) bolt locking devices	Stainless steel; nickel alloy	Reactor coolant and neutron flux	Loss of material due to wear; changes in dimensions due to void swelling or distortion	Chapter XI.M16A, “PWR Vessel Internals”	No
IV.B4.RP-260	IV.B4-31 (R-205)	Lower grid fuel assembly: (a) pads; (b) pad-to-rib section welds; (c) alloy X-750 dowels, cap screws and locking devices	Stainless steel; nickel alloy	Reactor coolant and neutron flux	Loss of fracture toughness due to neutron irradiation embrittlement	Chapter XI.M16A, “PWR Vessel Internals"	No

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Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
IV.B4.RP-260a		Lower grid fuel assembly: (a) pads; (b) pad-to-rib section welds; (c) alloy X-750 dowels, cap screws and locking devices	Stainless steel; nickel alloy	Reactor coolant and neutron flux	Cracking due to stress corrosion cracking or fatigue	Chapter XI.M16A, “PWR Vessel Internals” and Chapter XI.M2, “Water Chemistry” (for SCC mechanisms only)	No
IV.B4.RP-262	IV.B4-32 (R-203)	Lower grid assembly: alloy X-750 dowel-to-lower fuel assembly support pad locking welds	Nickel alloy	Reactor coolant and neutron flux	Cracking due to stress corrosion cracking	Chapter XI.M16A, “PWR Vessel Internals” and Chapter XI.M2, “Water Chemistry”	No
IV.B4.RP-261	IV.B4-32 (R-203)	Lower grid assembly: alloy X-750 dowel-to-guide block welds	Nickel alloy	Reactor coolant and neutron flux	Cracking due to stress corrosion cracking	Chapter XI.M16A, “PWR Vessel Internals” and Chapter XI.M2, “Water Chemistry”	No
IV.B4.R-53	IV.B4-37 (R-53)	Reactor vessel internal components	Stainless steel; nickel alloy	Reactor coolant and neutron flux	Cumulative fatigue damage due to fatigue	Fatigue is a time-limited aging analysis (TLAA) to be evaluated for the period of extended operation. See the SRP, Section 4.3 “Metal Fatigue,” for acceptable methods for meeting the requirements of 10 CFR 54.21(c)(1).	Yes, TLAA
IV.B4.RP-24	IV.B4-38 (RP-24)	Reactor vessel internal components	Stainless steel; nickel alloy	Reactor coolant and neutron flux	Loss of material due to pitting and crevice corrosion	Chapter XI.M2, “Water Chemistry”	No

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B4 Reactor Vessel Internals (PWR) – Babcock & Wilcox

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
IV.B4.RP-376		Reactor vessel internal components	Stainless steel; nickel alloy	Reactor coolant and neutron flux	Reduction in ductility and fracture toughness due to neutron irradiation	Ductility - Reduction in Fracture Toughness is a TLAA (BAW-2248A) to be evaluated for the period of extended operation. See the SRP, Section 4.7, "Other Plant-Specific TLAAs," for acceptable methods for meeting the requirements of 10 CFR 54.21(c)(1).	Yes, TLAA
IV.B4.RP-382	IV.B4-42 (R-179)	Reactor vessel internals: ASME Section XI, Examination Category B-N-3 core support structure components	Stainless steel; nickel alloy	Reactor coolant and neutron flux	Cracking due to fatigue, stress corrosion cracking, or irradiation-assisted stress corrosion cracking; loss of material due to wear	Chapter XI.M1, “ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD” or Chapter XI.M16A, “PWR Vessel Internals,” by invoking applicable 10 CFR 50.55a and ASME Section XI inservice inspection requirements	No
IV.B4.RP-352		Upper grid assembly: alloy X-750 dowel-to-upper fuel assembly support pad welds (all plants except Davis-Besse)	Nickel alloy	Reactor coolant and neutron flux	Cracking due to stress corrosion cracking	Chapter XI.M16A, “PWR Vessel Internals” and Chapter XI.M2, “Water Chemistry”	No

IV B4	REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM Reactor Vessel Internals (PWR) – Babcock & Wilcox						
Item	**Link**	**Structure and/or Component**	**Material**	**Environment**	**Aging Effect/ Mechanism**	**Aging Management Program (AMP)**	**Further Evaluation**
IV.B4.RP-236		Reactor internal "No Additional Measures" components	Stainless steel; nickel alloy	Reactor coolant and neutron flux	No additional aging management for reactor internal "No Additional Measures" components unless required by ASME Section XI, Examination Category B-N-3 or relevant operating experience exists	Chapter XI.M16A, "PWR Vessel Internals"	No
IV.B4.RP-400		Core support shield assembly: upper (top) flange weld	Stainless steel	Reactor coolant and neutron flux	Cracking due to stress corrosion cracking	Chapter XI.M16A, "PWR Vessel Internals" and Chapter XI.M2, "Water Chemistry"	No
IV.B4.RP-401		Core support shield assembly: upper (top) flange weld	Stainless steel	Reactor coolant and neutron flux	Loss of fracture toughness due to neutron irradiation embrittlement	Chapter XI.M16A, "PWR Vessel Internals"	No

(5) Revised version of GALL Report Chapter IX.C and IX.G

IX.C Selected Definitions & Use of Terms for Describing and Standardizing MATERIALS

Stainless steel	Products grouped under the term “stainless steel” include austenitic, ferritic, martensitic, precipitation-hardened (PH), or duplex stainless steel (Cr content >11%). These stainless steels may be fabricated using a wrought or cast process. These materials are susceptible to a variety of aging effects and mechanisms, including loss of material due to pitting and crevice corrosion, and cracking due to stress corrosion cracking. In some cases, when an aging effect is applicable to all of the various stainless steel categories, it can be assumed that the term “stainless steel” in the “Material” column of an AMR line-item in the GALL Report encompasses all stainless steel types. Cast austenitic stainless steel (CASS) is quite susceptible to loss of fracture toughness due to thermal and neutron irradiation embrittlement. In addition, MRP-227-A indicates that PH stainless steels or martensitic stainless steels may be susceptible to loss of fracture toughness by a thermal aging mechanism. Therefore, when loss of fracture toughness due to thermal and neutron irradiation embrittlement is an applicable aging effect and mechanism for a component in the GALL Report, the CASS, PH stainless steel, or martensitic stainless steel designation is specifically identified in an AMR line-item. Steel with stainless steel cladding also may be considered stainless steel when the aging effect is associated with the stainless steel surface of the material, rather than the composite volume of the material. Examples of stainless steel designations that comprise this category include A-286, SA193-Gr. B8, SA193-Gr. B8M, Gr. 660 (A-286), SA193-6, SA193-Gr. B8 or B-8M, SA453, Type 416, Type 403, 410, 420, and 431 martensitic stainless steels, Type 15-5, 17-4, and 13-8-Mo PH stainless steels, and SA-193, Grade B8 and B8M bolting materials. Examples of wrought austenitic stainless materials that comprise this category include Type 304, 304NG, 304L, 308, 308L, 309, 309L, 316 and 347. Examples of CASS that comprise this category include CF3, CF3M, CF8 and CF8M. [Ref. 6, 7, 30]

IX.G References

30. Welding Handbook, Seventh Edition, Volume 4, Metals and Their Weldability, American Welding Society, 1984, p. 76-145.

Appendix A, Section 2 – Revised version of the SRP-LR

(1) Revised version of SRP-LR Table 3.0-1

Table 3.0-1 FSAR Supplement for Aging Management of Applicable Systems				
GALL Chapter	**GALL Program**	**Description of Program**	**Implementation Schedule**	**Applicable GALL Report and SRP-LR Chapter References**
XI.M16A	PWR Vessel Internals	The program relies on implementation of the inspection and evaluation guidelines in EPRI Technical Report No. 1022863 (MRP-227-A) and EPRI Technical Report No. 1016609 (MRP-228) to manage the aging effects on the reactor vessel internal components. This program is used to manage (a) cracking, including stress corrosion cracking, primary water stress corrosion cracking, irradiation-assisted stress corrosion cracking, and cracking due to fatigue/cyclical loading; (b) loss of material induced by wear; (c) loss of fracture toughness due to either thermal aging, neutron irradiation embrittlement, or void swelling; (d) dimensional changes due to void swelling or distortion; and (e) loss of preload due to thermal and irradiation-enhanced stress relaxation or creep.	Program should be implemented prior to period of extended operation	GALL IV / SRP 3.1

(2) Revised version of SRP-LR Section 3.1.2, "Acceptance Criteria"

3.1.2.2.9 *Removed as a result of LR-ISG-2011-04*
3.1.2.2.10 *Removed as a result of LR-ISG-2011-04*
3.1.2.2.12 *Removed as a result of LR-ISG-2011-04*
3.1.2.2.13 *Removed as a result of LR-ISG-2011-04*
3.1.2.2.14 *Removed as a result of LR-ISG-2011-04*

(3) Revised version of SRP-LR Section 3.1.3, "Review Procedures"

3.1.3.2.9 *Removed as a result of LR-ISG-2011-04*
3.1.3.2.10 *Removed as a result of LR-ISG-2011-04*
3.1.3.2.12 *Removed as a result of LR-ISG-2011-04*
3.1.3.2.13 *Removed as a result of LR-ISG-2011-04*
3.1.3.2.14 *Removed as a result of LR-ISG-2011-04*

(5) Revised version of SRP-LR 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 Report

ID	Type	Component	Aging Effect/Mechanism	Aging Management Programs	Further Evaluation Recommended	Rev2 Item	Rev1 Item
3	BWR/ PWR	Stainless steel or nickel alloy reactor vessel internal components exposed to reactor coolant and neutron flux	Cumulative fatigue damage due to fatigue	Fatigue is a TLAA evaluated for the period of extended operation (See SRP, Section 4.3 "Metal Fatigue," for acceptable methods to comply with 10 CFR 54.21(c)(1)	Yes, TLAA (See subsection 3.1.2.2.1)	IV.B1.R-53 IV.B2.RP-303 IV.B3.RP-339 IV.B4.R-53	IV.B1-14 (R-53) IV.B2-31 (R-53) IV.B3-24 (R-53) IV.B4-37 (R-53)
15	PWR	Stainless steel Babcock & Wilcox (including CASS, martensitic SS, and PH SS) and nickel alloy reactor vessel internal components exposed to reactor coolant and neutron flux	Reduction of ductility and fracture toughness due to neutron irradiation embrittlement, and for CASS, martensitic SS, and PH SS due to thermal aging embrittlement	Ductility - Reduction in fracture toughness is a TLAA to be evaluated for the period of extended operation, See the SRP, Section 4.7, "Other Plant-Specific TLAAs," for acceptable methods of meeting the requirements of 10 CFR 54.21(c).	Yes, TLAA (See subsection 3.1.2.2.3.3)	IV.B4.RP-376	N/A
28	PWR	Stainless steel Combustion Engineering "Existing Programs" components exposed to reactor coolant and neutron flux	Loss of material due to wear; cracking due to stress corrosion cracking, irradiation-assisted stress corrosion cracking, or fatigue	Chapter XI.M16A, "PWR Vessel Internals," and Chapter XI.M2, "Water Chemistry" (for SCC mechanisms only)	No	IV.B3.RP-400	N/A
32	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, or loss of material due to wear	Chapter XI.M1, "ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD" or Chapter XI.M16A, "PWR Vessel Internals," invoking applicable 10 CFR 50.55a and ASME Section XI inservice inspection requirements for these components	No	IV.B2.RP-382 IV.B3.RP-382 IV.B4.RP-382	IV.B2-26 (R-142) IV.B3-22 (R-170) IV.B4-42 (R-179)

Table 3.1-1 Summary of Aging Management Programs for Reactor Vessel, Internals, and Reactor Coolant System Evaluated in Chapter IV of the GALL Report							
ID	**Type**	**Component**	**Aging Effect/Mechanism**	**Aging Management Programs**	**Further Evaluation Recommended**	**Rev2 Item**	**Rev1 Item**
51a	PWR	Stainless steel or nickel alloy Babcock & Wilcox reactor internal "Primary" components exposed to reactor coolant and neutron flux	Cracking due to stress corrosion cracking, irradiation-assisted stress corrosion cracking, or fatigue	Chapter XI.M16A, "PWR Vessel Internals," and Chapter XI.M2, "Water Chemistry" (for SCC mechanisms only)	No	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-256 IV.B4.RP-256a IV.B4.RP-258a IV.B4.RP-259a IV.B4.RP-261 IV.B4.RP-400	IV.B4-7 (R-125) N/A N/A IV.B4-13 (R-194) N/A IV.B4-25 (R-210) N/A N/A N/A IV.B4-25 (R-210) N/A N/A N/A IV.B4-32 (R-203) N/A
51b	PWR	Stainless steel or nickel alloy Babcock & Wilcox reactor internal "Expansion" components exposed to reactor coolant and neutron flux	Cracking due to stress corrosion cracking, irradiation-assisted stress corrosion cracking, fatigue, or overload	Chapter XI.M16A, "PWR Vessel Internals," and Chapter XI.M2, "Water Chemistry" (for SCC mechanisms only)	No	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-254 IV.B4.RP-254a IV.B4.RP-260a IV.B4.RP-262 IV.B4.RP-352 IV.B4.RP-250a IV.B4.RP-375	IV.B4-7 (R-125) N/A IV.B4-13 (R-194) N/A IV.B4-12 (R-196) N/A IV.B4-25 (R-210) N/A N/A IV.B4-32 (R-203) N/A N/A N/A
52a	PWR	Stainless steel or nickel alloy Combustion Engineering reactor internal "Primary" components exposed to reactor coolant and neutron flux	Cracking due to stress corrosion cracking, irradiation-assisted stress corrosion cracking, or fatigue	Chapter XI.M16A, "PWR Vessel Internals," and Chapter XI.M2, "Water Chemistry" (for SCC mechanisms only)	No	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-342 IV.B3.RP-358 IV.B3.RP-362a IV.B3.RP-363 IV.B3.RP-338 IV.B3.RP-343	IV.B3-2 (R-149) IV.B3-9 (R-162) N/A N/A N/A IV.B3-15 (R-155) IV.B3-15 (R-155) N/A N/A N/A N/A N/A N/A

Table 3.1-1 Summary of Aging Management Programs for Reactor Vessel, Internals, and Reactor Coolant System Evaluated in Chapter IV of the GALL Report							
ID	**Type**	**Component**	**Aging Effect/Mechanism**	**Aging Management Programs**	**Further Evaluation Recommended**	**Rev2 Item**	**Rev1 Item**
52b	PWR	Stainless steel or nickel alloy Combustion Engineering reactor internal "Expansion" components exposed to reactor coolant and neutron flux	Cracking due to stress corrosion cracking, irradiation-assisted stress corrosion cracking, or fatigue	Chapter XI.M16A, "PWR Vessel Internals," and Chapter XI.M2, "Water Chemistry" (for SCC mechanisms only)	No	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	NA IV.B3-9 (R-162) N/A N/A IV.B3-12 (R-155) IV.B3-23 (R-167) N/A IV.B3-23 (R-167) N/A
52c	PWR	Stainless steel or nickel alloy Combustion Engineering reactor internal "Existing Programs" components exposed to reactor coolant and neutron flux	Cracking due to stress corrosion cracking, irradiation-assisted stress corrosion cracking, or fatigue	Chapter XI.M16A, "PWR Vessel Internals," and Chapter XI.M2, "Water Chemistry" (for SCC mechanisms only)	No	IV.B3.RP-320 IV.B3.RP-334	IV.B3-9 (R-162) IV.B3-23 (R-167)
53a	PWR	Stainless steel or nickel alloy Westinghouse reactor internal "Primary" components exposed to reactor coolant and neutron flux	Cracking due to stress corrosion cracking, irradiation-assisted stress corrosion cracking, or fatigue	Chapter XI.M16A, "PWR Vessel Internals," and Chapter XI.M2, "Water Chemistry" (for SCC mechanisms only)	No	IV.B2.RP-270a IV.B2.RP-271 IV.B2.RP-275 IV.B2.RP-276 IV.B2.RP-280 IV.B2.RP-298 IV.B2.RP-302 IV.B2.RP-387	N/A IV.B2-10 (R-125) IV.B2-6 (R-128) IV.B2-8 (R-120) IV.B2-8 (R-120) IV.B2-28 (R-118) N/A N/A
53b	PWR	Stainless steel Westinghouse reactor internal "Expansion" components exposed to reactor coolant and neutron flux	Cracking due to stress corrosion cracking, irradiation-assisted stress corrosion cracking, or fatigue	Chapter XI.M16A, "PWR Vessel Internals," and Chapter XI.M2, "Water Chemistry" (for SCC mechanisms only)	No	IV.B2.RP-273 IV.B2.RP-278 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-387a	IV.B2-10 (R-125) IV.B2-8 (R-120) IV.B2-16 (R-133) IV.B2-24 (R-138) N/A N/A IV.B2-24 (R-138) IV.B2-24 (R-138) N/A
53c	PWR	Stainless steel or nickel alloy Westinghouse reactor internal "Existing Programs" components exposed to reactor coolant and neutron flux	Cracking due to stress corrosion cracking, irradiation-assisted stress corrosion cracking, or fatigue	Chapter XI.M16A, "PWR Vessel Internals," and Chapter XI.M2, "Water Chemistry" (for SCC mechanisms only)	No	IV.B2.RP-289 IV.B2.RP-301 IV.B2.RP-345 IV.B2.RP-346 IV.B2.RP-399 IV.B2.RP-355	IV.B2-20 (R-130) IV.B2-40 (R-112) N/A N/A N/A N/A

Table 3.1-1 Summary of Aging Management Programs for Reactor Vessel, Internals, and Reactor Coolant System Evaluated in Chapter IV of the GALL Report							
ID	**Type**	**Component**	**Aging Effect/Mechanism**	**Aging Management Programs**	**Further Evaluation Recommended**	**Rev2 Item**	**Rev1 Item**
54	PWR	Stainless steel bottom mounted instrument system flux thimble tubes (with or without chrome plating) exposed to reactor coolant and neutron flux (Westinghouse "Existing Programs" components)	Loss of material due to wear	Chapter XI.M16A, "PWR Vessel Internals," or Chapter XI.M37, "Flux Thimble Tube Inspection"	No	IV.B2.RP-284	IV.B2-13 (R-145)
55a	PWR	Stainless steel or nickel alloy Babcock and Wilcox reactor internal "No Additional Measures" components exposed to reactor coolant and neutron flux	No additional aging management for reactor internal "No Additional Measures" components unless required by ASME Section XI, Examination Category B-N-3 or relevant operating experience exists	Chapter XI.M16A, "PWR Vessel Internals"	No	IV.B4.RP-236	NA
55b	PWR	Stainless steel or nickel alloy Combustion Engineering reactor internal "No Additional Measures" components exposed to reactor coolant and neutron flux	No additional aging management for reactor internal "No Additional Measures" components unless required by ASME Section XI, Examination Category B-N-3 or relevant operating experience exists	Chapter XI.M16A, "PWR Vessel Internals"	No	IV.B3.RP-306	NA
55c	PWR	Stainless steel or nickel alloy Westinghouse reactor internal "No Additional Measures" components exposed to reactor coolant and neutron flux	No additional aging management for reactor internal "No Additional Measures" components unless required by ASME Section XI, Examination Category B-N-3 or relevant operating experience exists	Chapter XI.M16A, "PWR Vessel Internals"	No	IV.B2.RP-265	NA
56a	PWR	Stainless steel (SS, including CASS, PH SS or martensitic SS) or nickel alloy Combustion	Loss of fracture toughness due to neutron irradiation embrittlement and for CASS, martensitic SS, and PH SS	Chapter XI.M16A, "PWR Vessel Internals"	No	IV.B3.RP-315 IV.B3.RP-318 IV.B3.RP-359 IV.B3.RP-360	IV.B3-7 (R-165) IV.B3-8 (R-163) N/A N/A

Table 3.1-1 Summary of Aging Management Programs for Reactor Vessel, Internals, and Reactor Coolant System Evaluated in Chapter IV of the GALL Report							
ID	**Type**	**Component**	**Aging Effect/Mechanism**	**Aging Management Programs**	**Further Evaluation Recommended**	**Rev2 Item**	**Rev1 Item**
		Engineering reactor internal "Primary" components exposed to reactor coolant and neutron flux	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			IV.B3.RP-362 IV.B3.RP-364 IV.B3.RP-366 IV.B3.RP-365 IV.B3.RP-326	N/A N/A N/A N/A N/A
56b	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; 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	Chapter XI.M16A, "PWR Vessel Internals"	No	IV.B3.RP-317 IV.B3.RP-331 IV.B3.RP-359a IV.B3.RP-361 IV.B3.RP-362b	IV.B3-7 (R-165) N/A N/A N/A N/A
56c	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; 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	Chapter XI.M16A, "PWR Vessel Internals"	No	IV.B3.RP-319 IV.B3.RP-332 IV.B3.RP-334a IV.B3.RP-336 IV.B3.RP-357	IV.B3-9 (R-162) IV.B3-17 (R-156) N/A IV.B3-22 (R-170) N/A
58a	PWR	Stainless steel (SS, including CASS, PH SS or martensitic SS) or 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	Chapter XI.M16A, "PWR Vessel Internals"	No	IV.B4.RP-240 IV.B4.RP-240a IV.B4.RP-242 IV.B4.RP-247b IV.B4.RP-248b IV.B4.RP-249 IV.B4.RP-251 IV.B4.RP-251a IV.B4.RP-252	IV.B4-1 (R-128) N/A IV.B4-4 (R-183) N/A N/A IV.B4-12 (R-196) IV.B4-15 (R-190) N/A IV.B4-16 (R-188)

Table 3.1-1 Summary of Aging Management Programs for Reactor Vessel, Internals, and Reactor Coolant System Evaluated in Chapter IV of the GALL Report							
ID	**Type**	**Component**	**Aging Effect/Mechanism**	**Aging Management Programs**	**Further Evaluation Recommended**	**Rev2 Item**	**Rev1 Item**
			of preload due to wear; or loss of material due to wear			IV.B4.RP-254b IV.B4.RP-256b IV.B4.RP-258 IV.B4.RP-259 IV.B4.RP-401	N/A N/A IV.B4-4 (R-183) IV.B4-31 (R-205) N/A
58b	PWR	Stainless steel (SS, including CASS, PH SS or martensitic SS) or 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	Chapter XI.M16A, "PWR Vessel Internals"	No	IV.B4.RP-245b IV.B4.RP-246b IV.B4.RP-254b IV.B4.RP-260 IV.B4.RP-243 IV.B4.RP-243a IV.B4.RP-250 IV.B4.RP-375a	N/A N/A N/A IV.B4-31 (R-205) IV.B4-1 (R-128) N/A IV.B4-12 (R-196) N/A
59a	PWR	Stainless steel (SS, including CASS, PH SS or martensitic SS) or nickel alloy Westinghouse 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 thermal and irradiation enhanced stress relaxation or creep; or loss of material due to wear	Chapter XI.M16A, "PWR Vessel Internals"	No	IV.B2.RP-270 IV.B2.RP-272 IV.B2.RP-296 IV.B2.RP-297 IV.B2.RP-302a IV.B2.RP-354 IV.B2.RP-388 IV.B2.RP-300	IV.B2-1 (R-124) IV.B2-6 (R-128) N/A N/A N/A N/A N/A N/A
59b	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; 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	Chapter XI.M16A, "PWR Vessel Internals"	No	IV.B2.RP-274 IV.B2.RP-278a 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-388a	IV.B2-6 (R-128) N/A IV.B2-17 (R-135) IV.B2-21 (R-140) N/A N/A IV.B2-21 (R-140) IV.B2-22 (R-141) N/A

Table 3.1-1 Summary of Aging Management Programs for Reactor Vessel, Internals, and Reactor Coolant System Evaluated in Chapter IV of the GALL Report							
ID	**Type**	**Component**	**Aging Effect/Mechanism**	**Aging Management Programs**	**Further Evaluation Recommended**	**Rev2 Item**	**Rev1 Item**
			loss of material due to wear				
59c	PWR	Stainless steel (SS, including CASS, PH SS or martensitic SS) or nickel alloy 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; 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	Chapter XI.M16A, "PWR Vessel Internals"	No	IV.B2.RP-285 IV.B2.RP-288 IV.B2.RP-299 IV.B2.RP-356	IV.B2-14 (R-137) IV.B2-18 (R-132) IV.B2-34 (R-115) N/A

Appendix B

MARK-UP OF CHANGES TO THE GALL REPORT AND SRP-LR

Appendix B, Section 1 – Mark-up of Changes to the GALL Report

In the mark-up, strikethrough text indicates a deletion and underline text indicates an insertion. Double strikethrough text indicates the original location of the moved text and a double underline text indicates the final location of the moved text.

(1) Mark-up of changes to GALL Report AMP XI.M16A

XI.M16A PWR VESSEL INTERNALS

Program Description

This program relies on implementation of the Electric Power Research Institute (EPRI) Technical Report No. ~~1016596~~1022863, "Materials Reliability Program: Pressurized Water Reactor (PWR) Internals Inspection and Evaluation Guidelines," (MRP-227-A) and EPRI Report Technical No. 1016609, "Materials Reliability Program: Inspection Standard for PWR Internals," (MRP-228) to manage the aging effects on the pressurized water reactor (PWR) reactor vessel internal (RVI) components. The recommended activities in MRP-227-A and additional plant-specific activities not defined in MRP-227-A are implemented in accordance with Nuclear Energy Institute (NEI) 03-08, "Guideline for the Management of Materials Issues." The staff approved the augmented inspection and evaluation (I&E) criteria for PWR RVI components in NRC Safety Evaluation (SE), Revision 1, on MRP-227 by letter dated December 16, 2011.

This program is used to manage the effects of age-related degradation mechanisms that are applicable in general to the PWR RVI components at the facility. These aging effects include~~-~~: (a) ~~various forms of~~ cracking, including stress ~~-~~corrosion cracking (SCC), ~~which also encompasses~~ primary water stress ~~-~~corrosion cracking (PWSCC), irradiation-assisted stress corrosion cracking (IASCC), ~~or~~and cracking due to fatigue/cyclic~~al~~ loading; (b) loss of material induced by wear; (c) loss of fracture toughness due to either thermal aging or 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.

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

~~The~~ MRP-227-A guidance for selecting RVI components for inclusion in the inspection sample is based on a four-step ranking process. Through this process, the ~~reactor internals~~RVIs for all three PWR designs were assigned to one of the following four groups: "Primary~~,~~," "Expansion~~,~~," "Existing Programs~~,~~," and "No Additional Measures~~ components~~." Definitions of each group are provided in "Generic Aging Lessons Learned Report" (GALL Report), Revision 2, Chapter IX.B.

The result of this four-step sample selection process is a set of "Primary" ~~I~~internals ~~C~~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 of "Expansion"

~~I~~internals ~~C~~component locations that are specified to expand the sample should the indications be more severe than anticipated.

The degradation effects in a third set of internals locations are deemed to be adequately managed by "Existing Programs," such as American Society of Mechanical Engineers (ASME) Code, Section XI,[11] Examination Category B-N-3 examinations of core support structures. A fourth set of internals locations are deemed to require "N~~n~~o ~~a~~Additional ~~m~~Measures." ~~As a result, the program typically identifies 5 to 15% of the RVI locations as Primary Component locations for inspections, with another 7 to 10% of the RVI locations to be inspected as Expansion Components, as warranted by the evaluation of the inspection results. Another 5 to 15% of the internals locations are covered by Existing Programs, with the remainder requiring no additional measures. This process thus uses appropriate component functionality criteria, age-related degradation susceptibility criteria, and failure consequence criteria to identify the components that will be inspected under the program in a manner that conforms to the sampling criteria for sampling-based condition monitoring programs in Section A.1.2.3.4 of NRC Branch Position RLSB-1. Consequently, the sample selection process is adequate to assure that the intended function(s) of the PWR reactor internal components are maintained during the period of extended operation.~~

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

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

~~The program also includes future industry operating experience as incorporated in periodic revisions to MRP-227. The program thus provides reasonable assurance for the long-term integrity and safe operation of reactor internals in all commercial operating U.S. PWR nuclear power plants.~~

~~Age-related degradation in the reactor internals is managed through an integrated program. Specific features of the integrated program are listed in the following ten program elements. Degradation due to changes in material properties (e.g., loss of fracture toughness) was considered in the determination of inspection recommendations and is managed by the requirement to use appropriately degraded properties in the evaluation of identified defects. The integrated program is implemented by the applicant through an inspection plan that is submitted to the NRC for review and approval with the application for license renewal.~~

Evaluation and Technical Basis

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

1. *Scope of Program:* The scope of the program includes all RVI components ~~at the [*as an administrative action item for the AMP, the applicant to fill in the name of the applicant's nuclear facility, including applicable units*], which [*is/are*] built to a [*applicant to fill in Westinghouse, CE, or B&W, as applicable*]~~ based on the plant's applicable nuclear steam supply system ~~NSSS~~ design. The scope of the program applies the methodology and guidance in MRP-227-A~~the most recently NRC-endorsed version of MRP-227~~, which provides an augmented inspection and flaw evaluation methodology for assuring the functional integrity of safety-related internals in commercial operating U.S. PWR nuclear power plants designed by Babcock & Wilcox (B&W~~,~~), Combustion Engineering (CE~~,~~), and Westinghouse. The scope of components considered for inspection ~~under~~in MRP-227~~ guidance includes~~ A include core support structures~~ (typically denoted as Examination Category B-N-3 by the ASME Code, Section XI),~~, those RVI components that serve an intended license renewal safety function pursuant to criteria in 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.

The scope of the program does not include consumable items, such as fuel assemblies, reactivity control assemblies, and nuclear instrumentation~~, because these components are not typically within the scope of the components that are required to be subject to an aging management review (AMR), as defined by the criteria set in 10 CFR 54.21(a)(1).~~. The scope of the program also does not include welded attachments to the internal surface of the reactor vessel because these components are considered to be ASME Code Class 1 appurtenances to the reactor vessel and are ~~adequately~~ managed in accordance with an applicant's AMP that corresponds to GALL AMP XI.M1, "ASME Code, Section XI Inservice Inspection, Subsections IWB, IWC, and IWD."

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

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

2. *Preventive Actions:* ~~The guidance in~~ MRP-227-A 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~~. The program~~

~~description, evaluation, and technical basis of water chemistry are presented~~ in GALL AMP XI.M2, "Water Chemistry."

3. *Parameters Monitored/Inspected:* The program manages the following age-related degradation effects and mechanisms that are applicable in general to ~~the~~ RVI components at the facility: (a)~~-~~ cracking induced by SCC, PWSCC, IASCC, or fatigue/cyclic~~al~~ loading; (b)~~-~~ loss of material induced by wear; (c)~~-~~ loss of fracture toughness induced by either thermal aging or neutron irradiation embrittlement; (d)~~-~~ changes in dimensions due to void swelling~~ and irradiation growth,~~ or distortion~~, or deflection~~; and (e)~~-~~ loss of preload ~~caused~~ due to~~by~~ thermal and irradiation~~-~~-enhanced stress relaxation or creep.

For the management of cracking, the program monitors for evidence of surface breaking linear discontinuities if a visual inspection technique is used as the non-destructi~~on~~ve 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~~, or by void swelling and irradiation growth; instead~~. 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 ~~by~~(2) applying applicable reduced fracture toughness properties in the flaw evaluations~~ if~~, in cases where cracking is detected in the components and is extensive enough to ~~warrant~~ necessitate a supplemental flaw growth or flaw tolerance evaluation~~ under the MRP-227 guidance or ASME Code, Section XI requirements.~~. The program uses physical measurements to monitor for any dimensional changes due to void swelling or~~, irradiation growth,~~ distortion~~, or deflection.~~.

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

Specifically, the program implements the parameters monitored/inspected criteria consistent with the applicable tables in Section 4, "Aging Management Requirements," in MRP-227-A.

4. *Detection of Aging Effects:* ~~The detection of aging effects is covered in two places: (a) the guidance in Section 4 of MRP-227 provides an introductory discussion and justification of the examination~~ The inspection methods ~~selected for detecting the aging effects of interest; and (b) standards for examination~~are defined and established in Section 4 of MRP-227-A. Standards for implementing the inspection methods~~, procedures,~~ are defined and ~~personnel are provided~~established in ~~a companion document,~~ MRP-228. In all cases, well-established inspection methods are ~~were~~ selected. These methods include volumetric UT examination methods for detecting flaws in bolting~~, physical measurements for detecting changes in dimension,~~ 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). ~~The~~ 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~~ or affected assembly~~, 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.

~~In addition, the~~The program adopts the ~~recommended~~ guidance in MRP-227~~-A~~ for defining the "Expansion ~~criteria~~Criteria" that need to be applied to the inspection findings of "Primary ~~Components and Existing Requirement Components~~ " components and for expanding the examinations to include additional "Expansion~~ Components. As a result,~~" components. RVI component inspections ~~performed on the RVI components~~ are performed consistent with the inspection frequency and sampling bases for "Primary~~ Components,~~ " components, "Existing ~~Requirement Components~~Programs" components, and "Expansion~~ Components~~" components in MRP-227~~-A, which have been demonstrated to be in conformance with the inspection criteria, sampling basis criteria, and sample Expansion criteria in Section A.1.2.3.4 of NRC Branch Position RLSB-1~~.

~~Specifically, the program implements the parameters monitored/inspected criteria and bases for inspecting the relevant parameter conditions for [*as an administrative action item for the AMP, applicant is to select one of the following to finish the sentence, as applicable to its NSSS vendor for its internals: "B&W designed Primary Components in Table 4-1 of MRP-227"; "CE designed Primary Components in Table 4-2 of MRP-227;" or "Westinghouse designed Primary Components in Table 4-3 of MRP-227"*] and for [*as an administrative action item for the AMP, applicant is to select one of the following to finish the sentence, as applicable to its NSSS vendor for its internals: "for B&W designed Expansion Components in Table 4-4 of MRP-227;" "for CE designed expansion components in Table 4-5 of MRP-227;" and "for Westinghouse designed Expansion Components in Table 4-6 of MRP-227"*].~~

~~The program is supplemented by the following plant-specific Primary Component and Expansion Component inspections for the program (as applicable): [*As a relevant license renewal applicant action item, the applicant is to list (using criteria in MRP-227) each*~~

~~*additional RVI component that needs to be inspected as an additional plant-specific Primary Component for the applicant's program and each additional RVI component that needs to be inspected as an additional plant-specific Expansion Component for the applicant's program. For each plant specific component added as an additional primary or Expansion Component, the list should include the applicable aging effects that will be monitored for, the inspection method or methods used for monitoring, and the sample size and frequencies for the examinations*].~~

In~~ addition, in~~ some cases (as defined in MRP-227-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~~, deflection~~ or distortion.~~ The physical measurements methods applied in accordance with this program include [*Applicant to input physical measure methods identified by the MRP in response to NRC RAI No. 11 in the NRC's Request for Additional Information to Mr. Christen B. Larson, EPRI MRP on Topical Report MRP-227 dated November 12, 2009*].~~

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 MRP-227.

5. *Monitoring and Trending:* The methods for monitoring, recording, evaluating, and trending the data that result from the program's inspections are given in Section 6 of MRP-227-A and its subsections. ~~The~~ Flaw evaluation methods~~ include~~, 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. The ~~examinations~~examination and re-examinations ~~required by the~~that are implemented in accordance with MRP-227~~ guidance~~-A, together with the ~~requirements~~criteria specified in MRP-228 for inspection methodologies, inspection procedures, and inspection personnel, provide ~~timely detection, reporting, and corrective actions with respect to the effects of the age-related degradation mechanisms within the scope of the program. The extent of the examinations, beginning with the sample of susceptible PWR internals component locations identified as Primary Component locations, with the potential for inclusion of Expansion Component locations if the effects are greater than anticipated, plus the continuation of the Existing Programs activities, such as the ASME Code, Section XI, Examination Category B-N-3 examinations for core support structures, provides a high degree of confidence in the total~~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 a 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 to the applicable inspection technique and the resulting impact on the intended function(s) of the assembly containing the components.

6. *Acceptance Criteria:* Section 5 of MRP-227-A, which includes Table 5-1 for B&W-designed RVIs, Table 5-2 for CE-designed RVIs, and Table 5-3 for Westinghouse-designed RVIs, provides the specific examination and flaw evaluation acceptance criteria for the "Primary" and "Expansion ~~Component examinations. For~~ " RVI component examination methods. For RVI components addressed by examinations ~~referenced to~~ performed in accordance with the ASME Code, Section XI, the ~~IWB-3500~~ acceptance criteria ~~apply~~ in IWB-3500 are applicable. For ~~other~~RVI components covered by other "Existing Programs~~,~~," the ~~examination~~ acceptance criteria are described within the ~~Existing Programa~~applicable reference document.

~~The guidance in MRP-227 contains three types of examination~~ As applicable, the program establishes acceptance criteria~~:~~

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

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

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

7. *Corrective Actions:* ~~Corrective actions following the detection of unacceptable conditions are fundamentally provided for in each plant's corrective action program.~~ Any detected conditions that do not satisfy the examination acceptance criteria are required to be dispositioned through the plant corrective action program, which may require repair, replacement, or analytical evaluation for continued service until the next inspection. The disposition will ensure that design basis functions of the reactor internals components will continue to be fulfilled for all licensing basis loads and events. ~~Examples of methodologies that can be used to analytically disposition unacceptable conditions are found in the ASME Code, Section XI or in Section 6 of MRP-227. Section 6 of MRP-227 describes the options that are available for disposition of detected conditions that exceed the examination acceptance criteria of Section 5 of the report. These include engineering evaluation methods, as well as supplementary examinations to further characterize the detected condition, or the alternative of component repair and replacement procedures. The latter are subject to the requirements of the ASME Code, Section XI. The implementation of the guidance in MRP-227~~ The implementation of the guidance in MRP-227-A, 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 ~~action~~actions bases may be used to disposition relevant conditions if they have been previously approved or endorsed by the NRC. ~~Examples of previously NRC-endorsed alternative corrective actions bases include those corrective actions bases for Westinghouse-design RVI components that are defined in Tables 4-1, 4-2, 4-3, 4-4, 4-5, 4-6, 4-7 and 4-8 of Westinghouse Report No. WCAP-14577-Rev. 1-A, or for B&W-designed RVI components in B&W Report No. BAW-2248. Westinghouse Report No. WCAP-14577-Rev. 1-A was endorsed for use in an NRC SE to the Westinghouse Owners Group, dated February 10, 2001. B&W Report No. BAW-2248 was endorsed for use in an SE to Framatome Technologies on behalf of the B&W Owners Group, dated December 9, 1999.~~ Alternative corrective ~~action bases~~actions not approved or endorsed by the NRC will be submitted for NRC approval prior to their implementation.

8. *Confirmation Process:* Site quality assurance procedures, review and approval processes, and administrative controls are implemented in accordance with the recommendations of NEI 03-08 and the requirements of 10 CFR Part 50, Appendix B, or their equivalent, as applicable. ~~It is expected that the~~ The implementation of the guidance in MRP-227 ~~will provide~~-A, in conjunction with NEI 03-08 and other guidance documents, reports, or methodologies referenced in this AMP, provides an acceptable level of quality and an acceptable basis for ~~inspection~~confirming the quality of inspections, flaw ~~evaluation, and other elements of aging management of the PWR internals that are addressed in accordance with the 10 CFR Part 50, Appendix B, or their equivalent (as applicable), confirmation process, and administrative controls~~evaluations, and corrective actions.

9. *Administrative Controls:* The administrative controls for ~~such~~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. ~~Such~~ The evaluation in Section 3.5 of the NRC's SE, Revision1, on MRP-227 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 ~~program is thus expected to be established with a sufficient level of documentation and administrative controls to ensure effective long-term implementation~~licensee executive.

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

~~The application of the MRP-227 guidance will establish a considerable amount of~~ , including implementing procedures, are governed by NEI 03-08 and Appendix A of MRP-227-A. Consistent with MRP-227-A, the reporting of inspection results and operating experience ~~over the next few years. Section 7 of MRP-227 describes the reporting requirements for these applications, and the plan for evaluating the accumulated additional operating experience~~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 operating experience, as discussed in Appendix B of the GALL Report, which is documented in LR-ISG-2011-05.

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

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

~~NRC RAI No. 11 in the *NRC's Request for Additional Information* to the Mr. Christen B. Larson, EPRI MRP on Topical Report MRP-227 dated November 12, 2009.~~

~~NRC Safety Evaluation from C. I. Grimes [NRC] to R. A, Newton [Chairman, Westinghouse Owners Group], *Acceptance for Referencing of Generic License Renewal Program Topical Report Entitled "License Renewal Evaluation: Aging Management for Reactor Internals,"* WCAP-14577, Revision 1, February 10, 2001. (ADAMS Accession Number ML010430375).~~

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

~~NUREG-1800, Revision 2, Standard Review Plan for Review of License Renewal Applications for Nuclear Power Plants, Appendix A.1, "Aging Management Review - Generic (Branch Technical Position RLSB-1)," U.S. Nuclear Regulatory Commission, Washington, DC, 2010.~~

~~Westinghouse Non-Proprietary Class 3 Report No. WCAP-14577-Rev. 1-A, *License Renewal Evaluation: Aging Management for Reactor Internals*, Westinghouse Electric Company, Pittsburgh, PA [March 2001]. Report was submitted to the NRC Document Control Desk in a letter dated April 9, 2001. (ADAMS Accession Number ML011080790).~~

NRC Interim Staff Guidance LR-ISG-2011-05, *Ongoing Review Of Operating Experience*, March 16, 2012, (ADAMS Accession No. ML12044A215).

Nuclear Energy Institute (NEI) Report No. 03-08, Revision 2, *Guideline for the Management of Materials Issues*, ADAMS Accession No. ML101050334).

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(2) Mark-up of changes to GALL Report Chapter IV.B2

B2. REACTOR VESSEL INTERNALS (PWR) - WESTINGHOUSE

Systems, Structures, and Components

This section addresses the Westinghouse pressurized~~-~~water reactor (PWR) vessel internals~~ and consists of~~, which consist of components in the upper internals assembly, the control rod guide tube ~~assemblies~~assembly, the core barrel assembly, the baffle/former assembly, the lower ~~internal assembly, and the~~internals assembly, lower support assembly, thermal shield assembly, bottom mounted instrumentation ~~support structures. Based on Regulatory Guide 1.26, "Quality Group Classifications and Standards for Water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants," all structures and components that comprise the reactor vessel are governed by Group A or B Quality Standards.~~system, and alignment and interfacing components.

Common miscellaneous material/environment combinations where aging effects are not expected to degrade the ability of the structure or component to perform its intended function for the period of extended operation are included in IV.E.

System Interfaces

The systems that interface with the reactor vessel internals include the reactor pressure vessel (IV.A2).

~~Inspection Plan~~

~~An applicant will submit an inspection plan for reactor internals to the NRC for review and approval with the application for license renewal in accordance with Chapter XI.M16A, "PWR Vessel Internals."~~

IV REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM
B2 Reactor Vessel Internals (PWR) - Westinghouse

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
IV.B2.RP-300	IV.B2-33 (R-108)	Alignment and interfacing components: internals hold down spring	Stainless steel	Reactor coolant and neutron flux	Loss of preload due to thermal and irradiation enhanced stress relaxation; changes in dimensions due to void swelling or distortion; loss of material due to wear	Chapter XI.M16A, "PWR Vessel Internals" ~~Primary components (identified in the "Structure and Components" column)~~ ~~no Expansion components~~	No
IV.B2.RP-301	IV.B2-40 (R-112)	Alignment and interfacing components: upper core plate alignment pins	Stainless steel	Reactor coolant and neutron flux	Cracking due to stress~~-~~-corrosion cracking	~~'Chapter XI.M2, "Water Chemistry," and~~ Chapter XI.M16A, "PWR Vessel Internals" and Chapter XI.M2, "Water Chemistry" ~~Existing Program components (identified in the "Structure and Components" column)~~ ~~no Expansion components~~	No
IV.B2.RP-299	IV.B2-34 (R-115)	Alignment and interfacing components: upper core plate alignment pins	Stainless steel	Reactor coolant and neutron flux	Loss of material due to wear	Chapter XI.M16A, "PWR Vessel Internals" ~~Existing Program components (identified in the "Structure and Components" column)~~ ~~no Expansion components~~	No
IV.B2.RP-271	IV.B2-10 (R-125)	Baffle-to-former assembly: ~~accessible~~ baffle-to-former bolts	Stainless steel	Reactor coolant and neutron flux	Cracking due to irradiation-assisted stress~~-~~-corrosion cracking ~~and~~or fatigue	~~Chapter XI.M2, "Water Chemistry," and~~ Chapter XI.M16A, "PWR Vessel Internals" ~~Primary components (identified in the "Structure and Components" column)~~ and Chapter XI.M2, "Water Chemistry" (for ~~Expansion components see AMR Items IV.B2.RP-273 and IV.B2.RP-286)~~SCC mechanisms only)	No

IV B2	REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM Reactor Vessel Internals (PWR) - Westinghouse						
Item	**Link**	**Structure and/or Component**	**Material**	**Environment**	**Aging Effect/ Mechanism**	**Aging Management Program (AMP)**	**Further Evaluation**
IV.B2.RP-272	IV.B2-6 (R-128)	Baffle-to-former assembly: ~~accessible~~ baffle-to-former bolts	Stainless steel	Reactor coolant and neutron flux	Loss of fracture toughness due to neutron irradiation embrittlement; ~~change~~changes in dimensions due to void swelling or distortion; loss of preload due to thermal and irradiation enhanced stress relaxation or creep	Chapter XI.M16A, "PWR Vessel Internals" ~~Primary components (identified in the "Structure and Components" column)~~ ~~(for Expansion components see AMR Items IV.B2.RP-274 and IV.B2.RP-287)~~	No
IV.B2.RP-270	IV.B2-1 (R-124)	Baffle-to-former assembly: baffle and former plates	Stainless steel	Reactor coolant and neutron flux	Changes in dimensions due to void swelling or distortion	Chapter XI.M16A, "PWR Vessel Internals" ~~Primary components (identified in the "Structure and Components" column)~~ ~~no Expansion components~~	No
IV.B2.RP-270a	IV.B2-1 (R-124)	Baffle-to-former assembly: baffle and former plates	Stainless steel	Reactor coolant and neutron flux	Cracking due to irradiation-assisted stress corrosion cracking	Chapter XI.M16A, "PWR Vessel Internals" and Chapter XI.M2, "Water Chemistry"	No
IV.B2.RP-275	IV.B2-6 (R-128)	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 stress~~-~~-corrosion cracking ~~and~~or fatigue	~~Chapter XI.M2, "Water Chemistry," and~~ Chapter XI.M16A, "PWR Vessel Internals" and Chapter XI.M2, "Water Chemistry" (for SCC mechanisms only) ~~Primary components (identified in the "Structure and Components" column)~~ ~~no Expansion components~~	No

IV B2	REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM Reactor Vessel Internals (PWR) - Westinghouse						
Item	**Link**	**Structure and/or Component**	**Material**	**Environment**	**Aging Effect/ Mechanism**	**Aging Management Program (AMP)**	**Further Evaluation**
IV.B2.RP-354		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	Chapter XI.M16A, "PWR Vessel Internals" ~~Primary components (identified in the "Structure and Components" column)~~ ~~no Expansion components~~	No
IV.B2.RP-273	IV.B2-10 (R-125)	Baffle-to-former assembly: barrel-to-former bolts	Stainless steel	Reactor coolant and neutron flux	Cracking due to irradiation-assisted stress~~-~~corrosion cracking ~~and~~or fatigue	~~Chapter XI.M2, "Water Chemistry," and~~ Chapter XI.M16A, "PWR Vessel Internals" ~~Expansion components (identified in the "Structure and Components" column)~~ and Chapter XI.M2, "Water Chemistry (for ~~Primary components see AMR Item IV.B2.RP-271)~~SCC mechanisms only)	No
IV.B2.RP-274	IV.B2-6 (R-128)	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	Chapter XI.M16A, "PWR Vessel Internals" ~~Expansion components (identified in the "Structure and Components" column)~~ ~~(for Primary components see AMR Item IV.B2.RP-272)~~	No

IV REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM
B2 Reactor Vessel Internals (PWR) - Westinghouse

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
IV.B2.RP-284	IV.B2-12 (R-143)	Bottom mounted instrument system: flux thimble tubes	Stainless steel (with or without chrome plating)	Reactor coolant and neutron flux	Loss of material due to wear	Chapter XI.M16A, "PWR Vessel Internals" ~~Existing Program components (identified in the "Structure and Components" column) No expansion components; and~~ or Chapter XI.M37~~, "~~,"Flux Thimble Tube Inspection~~"~~"	No
IV.B2.RP-293	IV.B2-24 (R-138)	Bottom-mounted instrumentation system: bottom-mounted instrumentation (BMI) column bodies	Stainless steel	Reactor coolant and neutron flux	Cracking due to fatigue	~~Chapter XI.M2, "Water Chemistry," and~~ Chapter XI.M16A, "PWR Vessel Internals" ~~Expansion components (identified in the "Structure and Components" column) (for Primary components see AMR Item IV.B2.RP-298)~~	No
IV.B2.RP-292	~~IV.B2-21(R-140)~~	Bottom-mounted instrumentation system: bottom-mounted ~~instrumentation~~instrument (BMI) column bodies	Stainless steel	Reactor coolant and neutron flux	Loss of fracture toughness due to neutron irradiation embrittlement	Chapter XI.M16A, "PWR Vessel Internals" ~~Expansion components (identified in the "Structure and Components" column) (for Primary components see AMR Item IV.B2.RP-297)~~	No
IV.B2.RP-296		Control rod guide tube (CRGT) assemblies: CRGT guide plates (cards)	Stainless steel	Reactor coolant and neutron flux	Loss of material due to wear	Chapter XI.M16A, "PWR Vessel Internals" ~~Primary Components (identified in the "Structure and Components" column) (for Expansion components see AMR Line Item IV.B2.RP-386)~~	No

IV B2	REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM Reactor Vessel Internals (PWR) - Westinghouse						
Item	**Link**	**Structure and/or Component**	**Material**	**Environment**	**Aging Effect/ Mechanism**	**Aging Management Program (AMP)**	**Further Evaluation**
IV.B2.RP-298	IV.B2-28 (R-118)	Control rod guide tube (CRGT) assemblies: CRGT lower flange welds ~~(accessible)~~	Stainless steel	Reactor coolant and neutron flux	Cracking due to stress~~-~~corrosion cracking ~~and~~or fatigue	~~Chapter XI.M2, "Water Chemistry," and~~ Chapter XI.M16A, "PWR Vessel Internals" ~~Primary components (identified in the "Structure and Components" column)~~ and Chapter XI.M2, "Water Chemistry" (for ~~Expansion components see AMR Items IV.B2.RP-291 and IV.B2.RP-293)~~SCC mechanisms only)	No
IV.B2.RP-297		Control rod guide tube (CRGT) assemblies: CRGT lower flange welds ~~(accessible)~~	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	Chapter XI.M16A, "PWR Vessel Internals" ~~Primary components (identified in the "Structure and Components" column)~~ ~~(for Expansion components see AMR Items IV.B2.RP-290 and IV.B2.RP-292)~~	No
~~IV.B2.RP-386~~		~~Control rod guide tube (CRGT) assemblies: C-tubes and sheaths~~	~~Stainless steel~~	~~Reactor coolant and neutron flux~~	~~Loss of material due to wear~~	~~Chapter XI.M16A, "PWR Vessel Internals"~~ ~~Expansion components (identified in the "Structure and Components" column) are only the components associated with a primary component that exceeded the acceptance limit.~~ ~~(for Primary components see AMR Item IV.B2.RP-296)~~	~~No~~
~~IV.B2.RP-355~~ IV.B2.RP-355		Control rod guide tube (CRGT) assemblies: guide tube support pins (split pins)	~~Nickel~~Stainless steel; nickel alloy	Reactor coolant and neutron flux	Cracking due to stress~~-~~corrosion cracking ~~and~~or fatigue	~~A plant-specific aging management program is to be evaluated~~Chapter XI.M16A, "PWR Vessel Internals" and Chapter XI.M2, "Water Chemistry" (for SCC mechanisms only)	~~Yes, plant-specific~~No

IV B2	REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM Reactor Vessel Internals (PWR) - Westinghouse						
Item	**Link**	**Structure and/or Component**	**Material**	**Environment**	**Aging Effect/ Mechanism**	**Aging Management Program (AMP)**	**Further Evaluation**
IV.B2.RP-356		Control rod guide tube (CRGT) assemblies: guide tube support pins (split pins)	~~Nickel~~Stainless steel; nickel alloy	Reactor coolant and neutron flux	Loss of material due to wear	~~A plant-specific aging management program is to be evaluated~~Chapter XI.M16A, "PWR Vessel Internals"	~~Yes, plant-specific~~No
IV.B2.RP-387		Core barrel assembly: upper core barrel ~~axial~~and lower core barrel circumferential (girth) welds	Stainless steel	Reactor coolant and neutron flux	Cracking due to stress~~-~~corrosion cracking~~, and~~ or irradiation-assisted stress -corrosion cracking or fatigue	~~Chapter XI.M2, "Water Chemistry," and~~ Chapter XI.M16A, "PWR Vessel Internals" ~~Expansion components (identified in the "Structure and Components" column)~~ and Chapter XI.M2, "Water Chemistry" (for ~~Primary components see AMR Item IV.B2.RP-276~~SCC mechanisms only)	No
IV.B2.RP-387a		Core barrel assembly: upper core barrel and lower core barrel vertical (axial) welds	Stainless steel	Reactor coolant and neutron flux	Cracking due to stress corrosion cracking or irradiation-assisted stress corrosion cracking or fatigue	Chapter XI.M16A, "PWR Vessel Internals" and Chapter XI.M2, "Water Chemistry" (for SCC mechanisms only)	No
IV.B2.RP-388		Core barrel assembly: upper core barrel ~~axial~~and lower core barrel circumferential (girth) welds	Stainless steel	Reactor coolant and neutron flux	Loss of fracture toughness due to neutron irradiation embrittlement	Chapter XI.M16A, "PWR Vessel Internals" ~~Expansion components (identified in the "Structure and Components" column)~~ ~~(for Primary components see AMR Item IV.B2.RP-276)~~	No
IV.B2.RP-~~282~~388a	~~IV.B2-8(R-120)~~	Core barrel assembly: upper core barrel ~~flange~~and lower core barrel vertical (axial) welds	Stainless steel	Reactor coolant and neutron flux	~~Cracking~~Loss of fracture toughness due to ~~stress corrosion cracking and fatigue~~neutron irradiation embrittlement	~~Chapter XI.M2, "Water Chemistry," and~~ Chapter XI.M16A, "PWR Vessel Internals" ~~Expansion components (identified in the "Structure and Components" column)~~ ~~(for Primary components see AMR Item IV.B2.RP-276)~~	No

IV REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM
B2 Reactor Vessel Internals (PWR) - Westinghouse

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
IV.B2.RP-345		Core barrel assembly: core barrel flange	Stainless steel	Reactor coolant and neutron flux	Loss of material due to wear	Chapter XI.M16A, "PWR Vessel Internals" ~~Existing Program components (identified in the "Structure and Components" column) no Expansion components~~	No
IV.B2.RP-278	IV.B2-8 (R-120)	Core barrel assembly: core barrel outlet nozzle welds	Stainless steel	Reactor coolant and neutron flux	~~Cracking due to stress corrosion cracking and fatigue~~ Cracking due to stress corrosion cracking or fatigue	~~Chapter XI.M2, "Water Chemistry," and~~ Chapter XI.M16A, "PWR Vessel Internals" ~~Expansion component (identified in the "Structure and Components" column)~~ and Chapter XI.M2, "Water Chemistry" (for ~~Primary components see AMR Item IV.B2.RP-276)~~SCC mechanisms only)	No
IV.B2.RP-~~280~~278a	~~IV.B2-8(R-120)~~	Core barrel assembly: ~~lower~~ core barrel ~~flange weld~~outlet nozzle welds	Stainless steel	Reactor coolant and neutron flux	~~Cracking due to stress corrosion cracking and irradiation-assisted stress corrosion cracking~~ Loss of fracture toughness due to neutron irradiation embrittlement	~~Chapter XI.M2, "Water Chemistry," and~~ Chapter XI.M16A, "PWR Vessel Internals" ~~Expansion component (identified in the "Structure and Components" column) (for Primary components see AMR Item IV.B2.RP-276)~~	No
IV.B2.RP-~~281~~280	IV.B2-~~98~~ (R-~~122~~120)	Core barrel assembly: lower core barrel flange weld	Stainless steel	Reactor coolant and neutron flux	~~Loss of fracture toughness~~Cracking due to ~~neutron irradiation embrittlement~~stress corrosion cracking or fatigue	Chapter XI.M16A, "PWR Vessel Internals" ~~Expansion Components (identified in the "Structure~~ and ~~Components" column)~~ Chapter XI.M2, "Water Chemistry" (for ~~Primary components see AMR Item IV.B2.RP-276)~~SCC mechanisms only)	No

IV REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM
B2 Reactor Vessel Internals (PWR) - Westinghouse

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
IV.B2.RP-276	IV.B2-8 (R-120)	Core barrel assembly: upper core barrel flange weld	Stainless steel	Reactor coolant and neutron flux	Cracking due to stress~~-~~corrosion cracking and irradiation-assisted stress corrosion cracking	~~Chapter XI.M2, "Water Chemistry," and~~ Chapter XI.M16A, "PWR Vessel Internals" and Chapter XI.M2, "Water Chemistry" ~~Primary components (identified in the "Structure and Components" column) (for Expansion components see AMR Items IV.B2.RP-278, IV.B2.RP-280, IV.B2.RP-282, IV.B2.RP-294, IV.B2.RP-295,IV.B2.RP-281, IV.B2.RP-387, and IV.B2.RP-388)~~	No
IV.B2.RP-285	IV.B2-14 (R-137)	Lower internals assembly: clevis insert bolts or screws	Nickel alloy	Reactor coolant and neutron flux	Loss of material ~~due to wear~~due to wear; loss of preload due to thermal and irradiation enhanced stress relaxation or creep	Chapter XI.M16A, "PWR Vessel Internals" ~~Existing Program components (identified in the "Structure and Components" column) no Expansion components~~	No
IV.B2.RP-399		Lower internals assembly: clevis insert bolts or screws	Stainless steel; nickel alloy	Reactor coolant and neutron flux	Cracking due to primary water stress corrosion cracking, irradiation-assisted stress corrosion cracking or fatigue	Chapter XI.M16A, "PWR Vessel Internals" and Chapter XI.M2, "Water Chemistry" (for SCC mechanisms only)	No
IV.B2.RP-289	IV.B2-20 (R-130)	Lower internals assembly: lower core plate and extra-long (XL) lower core plate	Stainless steel	Reactor coolant and neutron flux	Cracking due to irradiation-assisted stress~~-~~corrosion cracking~~,~~ ~~and~~ or fatigue	~~'Chapter XI.M2, "Water Chemistry," and~~ Chapter XI.M16A, "PWR Vessel Internals" and Chapter XI.M2, "Water Chemistry" (for SCC mechanisms only) ~~Existing Program components (identified in the "Structure and Components" column) no Expansion components~~	No

IV B2	REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM Reactor Vessel Internals (PWR) - Westinghouse						
Item	**Link**	**Structure and/or Component**	**Material**	**Environment**	**Aging Effect/ Mechanism**	**Aging Management Program (AMP)**	**Further Evaluation**
IV.B2.RP-288	IV.B2-18 (R-132)	Lower internals assembly: lower core plate and 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	Chapter XI.M16A, "PWR Vessel Internals" ~~Existing Program components (identified in the "Structure and Components" column) no Expansion components~~	No
IV.B2.RP-291	IV.B2-24 (R-138)	Lower support assembly: lower support column bodies (cast)	Cast austenitic stainless steel	Reactor coolant and neutron flux	Cracking due to irradiation-assisted stress~~-~~corrosion cracking	~~Chapter XI.M2, "Water Chemistry," and~~ Chapter XI.M16A, "PWR Vessel Internals" and Chapter XI.M2, "Water Chemistry" ~~Expansion components (identified in the "Structure and Components" column) (for Primary components see AMR Item IV.B2.RP-298)~~	No
IV.B2.RP-290	IV.B2-21 (R-140)	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	Chapter XI.M16A, "PWR Vessel Internals" ~~Expansion components (identified in the "Structure and Components" column) (for Primary components see AMR Item IV.B2.RP-297)~~	No
IV.B2.RP-291a		Lower support assembly: lower support forging or casting	Stainless steel	Reactor coolant and neutron flux	Cracking due to stress corrosion cracking or fatigue	Chapter XI.M16A, "PWR Vessel Internals" and Chapter XI.M2, "Water Chemistry" (for SCC mechanisms only)	No
IV.B2.RP-290a		Lower support assembly: lower support forging or casting	Stainless steel	Reactor coolant and neutron flux	Loss of fracture toughness due to neutron irradiation embrittlement (and thermal aging embrittlement for CASS, PH SS, and martensitic SS)	Chapter XI.M16A, "PWR Vessel Internals"	No

IV B2	REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM Reactor Vessel Internals (PWR) - Westinghouse						
Item	**Link**	**Structure and/or Component**	**Material**	**Environment**	**Aging Effect/ Mechanism**	**Aging Management Program (AMP)**	**Further Evaluation**
IV.B2.RP-294	IV.B2-24 (R-138)	Lower support assembly: lower support column bodies (non-cast)	Stainless steel	Reactor coolant and neutron flux	Cracking due to irradiation-assisted stress~~-~~corrosion cracking	~~Chapter XI.M2, "Water Chemistry," and~~ Chapter XI.M16A, "PWR Vessel Internals" and Chapter XI.M2, "Water Chemistry" ~~Expansion components (identified in the "Structure and Components" column) (for Primary components see AMR Item IV.B2.RP-276)~~	No
IV.B2.RP-295	~~IV.B2-22(R-141)~~	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	Chapter XI.M16A, "PWR Vessel Internals" ~~Expansion Components (identified in the "Structure and Components" column) (for Primary components see AMR Item IV.B2.RP-276)~~	No
IV.B2.RP-286	IV.B2-16 (R-133)	Lower support assembly: lower support column bolts	Stainless steel~~; nickel alloy~~	Reactor coolant and neutron flux	Cracking due to irradiation-assisted stress~~-~~corrosion cracking ~~and~~or fatigue	~~Chapter XI.M2, "Water Chemistry," and~~ Chapter XI.M16A, "PWR Vessel Internals" ~~Expansion components (identified in the "Structure and Components" column)~~ and Chapter XI.M2, "Water Chemistry" (for ~~Primary components see AMR Item IV.B2.RP-271)~~SCC mechanisms only)	No
IV.B2.RP-287	IV.B2-17 (R-135)	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	Chapter XI.M16A, "PWR Vessel Internals" ~~Expansion component (identified in the "Structure and Components" column) (for Primary components see AMR Item IV.B2.RP-272)~~	No

IV B2	REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM Reactor Vessel Internals (PWR) - Westinghouse						
Item	**Link**	**Structure and/or Component**	**Material**	**Environment**	**Aging Effect/ Mechanism**	**Aging Management Program (AMP)**	**Further Evaluation**
IV.B2.RP-303	IV.B2-31 (R-53)	Reactor vessel internal components	Stainless steel; nickel alloy	Reactor coolant and neutron flux	Cumulative fatigue damage due to fatigue	Fatigue is a time-limited aging analysis (TLAA) to be evaluated for the period of extended operation. See the SRP, Section 4.3 "Metal Fatigue," for acceptable methods for meeting the requirements of 10 CFR 54.21(c)(1).	Yes~~,~~ – TLAA
IV.B2.RP-24	IV.B2-32 (RP-24)	Reactor vessel internal components	Stainless steel; nickel alloy	Reactor coolant and neutron flux	Loss of material due to pitting and crevice corrosion	Chapter XI.M2, "Water Chemistry"	No
IV.B2.RP-~~268~~382	IV.B2-26 (R-142)	Reactor vessel ~~internal~~internals: ASME Section XI, Examination Category B-N-3 core support structure components (~~inaccessible locations)~~not already identified as "Existing Programs" components in MRP-227-A)	Stainless steel; nickel alloy	Reactor coolant and neutron flux	Cracking due to fatigue, stress corrosion cracking, ~~and~~or irradiation~~-~~ assisted stress corrosion cracking; loss of material due to wear	Chapter XI.~~M2, "Water Chemistry,"~~M1, "ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD" or Chapter XI.M16A, "PWR Vessel Internals"~~–~~," by invoking applicable 10 CFR 50.55a and ASME Section XI inservice inspection requirements	~~Yes, if accessible Primary, Expansion or Existing program components indicate aging effects that need management~~No
IV.B2.RP-302		Thermal shield assembly: thermal shield flexures	Stainless steel	Reactor coolant and neutron flux	Cracking due to fatigue	Chapter XI.M16A, "PWR Vessel Internals"	No

IV B2	REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM Reactor Vessel Internals (PWR) - Westinghouse						
Item	**Link**	**Structure and/or Component**	**Material**	**Environment**	**Aging Effect/ Mechanism**	**Aging Management Program (AMP)**	**Further Evaluation**
IV.B2.RP-~~269~~302a		~~Reactor vessel internal components (inaccessible locations)~~Thermal shield assembly: thermal shield flexures	Stainless steel~~; nickel alloy~~	Reactor coolant and neutron flux	Loss of ~~fracture toughness due to neutron irradiation embrittlement; change in dimension due to void swelling; loss of preload due to thermal and irradiation enhanced stress relaxation; loss of~~ material due to wear	Chapter XI.M16A, "PWR Vessel Internals"	~~Yes, if accessible Primary, Expansion or Existing program components indicate aging effects that need management~~No
IV.B2.RP-265		Reactor internal "No Additional Measures" components~~Reactor vessel internal components with no additional measures~~	Stainless steel; nickel alloy	Reactor coolant and neutron flux	No additional aging management for reactor internal "No Additional Measures" components unless required by ASME Section XI, Examination Category B-N-3 or relevant operating experience exists~~Cracking due to stress corrosion cracking, and irradiation-assisted stress corrosion cracking~~	~~Chapter XI.M2, "Water Chemistry," and~~ Chapter XI.M16A, "PWR Vessel Internals"~~ Note: Components with no additional measures are not uniquely identified in GALL tables - Components with no additional measures are defined in Section 3.3.1 of MRP-227, "Materials Reliability Program: Pressurized Water Reactor Internals Inspection and Evaluation Guidelines"~~	No
IV.B2.RP-~~267~~291b		~~Reactor vessel internal components with no additional measures~~ Upper Internals Assembly; upper core plate	Stainless steel~~; nickel alloy~~	Reactor coolant and neutron flux	~~Loss of fracture toughness~~ Cracking due to ~~neutron irradiation embrittlement; change in dimension due to void swelling; loss of preload due to thermal and irradiation enhanced stress relaxation~~fatigue~~; loss of material due to wear~~	Chapter XI.M16A, "PWR Vessel Internals" ~~Note: Components with no additional measures are not uniquely identified in GALL tables - Components with no additional measures are defined in Section 3.3.1 of MRP-227, "Materials Reliability Program: Pressurized Water Reactor Internals Inspection and Evaluation Guidelines"~~	No
~~IV.B2.RP-382~~	~~IV.B2-26(R-142)~~	~~Reactor vessel internals: core support structure~~	~~Stainless steel; nickel alloy; cast austenitic stainless steel~~	~~Reactor coolant and neutron flux~~	~~Cracking, or Loss of material due to wear~~	~~Chapter XI.M1, "ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD"~~	~~No~~

IV B2	REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM Reactor Vessel Internals (PWR) - Westinghouse						
Item	**Link**	**Structure and/or Component**	**Material**	**Environment**	**Aging Effect/ Mechanism**	**Aging Management Program (AMP)**	**Further Evaluation**
IV.B2.RP-~~302~~290b		Upper Internals Assembly; upper core plate~~Thermal shield assembly; thermal shield flexures~~	Stainless steel	Reactor coolant and neutron flux	~~Cracking due to fatigue; loss~~Loss of material due to wear	Chapter XI.M16A, "PWR Vessel Internals" ~~Primary components (identified in the "Structure and Components" column)~~ ~~no Expansion components~~	No
IV.B2.RP-346		Upper ~~i~~Internals ~~a~~Assembly: upper support ring or skirt	Stainless steel	Reactor coolant and neutron flux	Cracking due to stress corrosion cracking ~~and~~ or fatigue	~~'Chapter XI.M2, "Water Chemistry," and~~ Chapter XI.M16A, "PWR Vessel Internals" and Chapter XI.M2, "Water Chemistry" (for SCC mechanisms only) ~~Existing Program components (identified in the "Structure and Components" column)~~ ~~no Expansion components~~	No

(3) Mark-up of changes to GALL Report Chapter IV.B3

B3. REACTOR VESSEL INTERNALS (PWR) - COMBUSTION ENGINEERING

Systems, Structures, and Components

This section addresses the Combustion Engineering (CE) pressurized-water reactor (PWR) vessel internals and consists of, which consist of components in the upper internals assembly, the control element assembly (CEA) shrouds,), the core support barrel assembly, the core shroud assembly, and the lower internal assembly. Based on Regulatory Guide 1.26, "Quality Group Classifications and Standards for Water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants," all structures and components that comprise the reactor vessel are governed by Group A or B Quality Standards support structure assembly, and encore instrumentation (ICI) components.

Common miscellaneous material/environment combinations where aging effects are not expected to degrade the ability of the structure or component to perform its intended function for the period of extended operation are included in IV.E.

System Interfaces

The systems that interface with the reactor vessel internals include the reactor pressure vessel (IV.A2).

Inspection Plan

An applicant will submit an inspection plan for reactor internals to the NRC for review and approval with the application for license renewal in accordance with Chapter XI.M16A, "PWR Vessel Internals."

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

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
IV.B3.RP-312	IV.B3-2 (R-149)	Control Element Assembly (CEA): ~~shroud assemblies:~~ instrument guide tubes in peripheral CEA assemblies	Stainless steel	Reactor coolant and neutron flux	Cracking due to stress~~-~~ corrosion cracking ~~and~~or fatigue	~~Chapter XI.M2, "Water Chemistry," and~~ Chapter XI.M16A, "PWR Vessel Internals" ~~Primary components (identified in the "Structure and Components" column)~~ and Chapter XI.M2, "Water Chemistry" (for ~~Expansion components see AMR Item IV.B3.RP-313)~~SCC mechanisms only)	No
IV.B3.RP-313		Control Element Assembly (CEA): ~~shroud assemblies:~~ remaining instrument guide tubes in CEA assemblies	Stainless steel	Reactor coolant and neutron flux	Cracking due to stress~~-~~ corrosion cracking ~~and~~or fatigue	~~Chapter XI.M2, "Water Chemistry," and~~ Chapter XI.M16A, ~~"~~"PWR Vessel Internals~~"~~ ~~Expansion components (identified in the "Structure and Components" column)~~ " and Chapter XI.M2, "Water Chemistry" (for ~~Primary components see AMR Item IV.B3.RP-312)~~SCC mechanisms only)	No
IV.B3.RP-320	IV.B3-9 (R-162)	Core shroud assemblies (all plants): guide lugs ~~and~~; guide lug ~~insert~~inserts and bolts	Stainless steel	Reactor coolant and neutron flux	Cracking due to fatigue	~~'Chapter XI.M2, "Water Chemistry," and~~ Chapter XI.M16A, "PWR Vessel Internals" ~~Existing Program components (identified in the "Structure and Components" column) no Expansion components~~	No
IV.B3.RP-319	IV.B3-9 (R-162)	Core shroud assemblies (all plants): guide lugs ~~and~~; guide lug ~~insert~~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	Chapter XI.M16A, "PWR Vessel Internals~~"~~" ~~Existing Program components (identified in the "Structure and Components" column) no Expansion components~~	No

IV B3	REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM Reactor Vessel Internals (PWR) - Combustion Engineering						
Item	**Link**	**Structure and/or Component**	**Material**	**Environment**	**Aging Effect/ Mechanism**	**Aging Management Program (AMP)**	**Further Evaluation**
IV.B3.RP-358		Core shroud assemblies (for bolted core shroud assemblies): ~~(a)~~assembly components, including shroud plates and ~~(b)~~ former plates	Stainless steel	Reactor coolant and neutron flux	Cracking due to irradiation-assisted stress~~-~~ corrosion cracking	~~Chapter XI.M2, "Water Chemistry," and~~ Chapter XI.M16A, "PWR Vessel Internals" and Chapter XI.M2, "Water Chemistry" ~~Expansion components (identified in the "Structure and Components" column) (for Primary component see AMR Item IV.B3.RP-314)~~	No
IV.B3.RP-318	IV.B3-8 (R-163)	Core shroud assemblies (for bolted core shroud assemblies): ~~(a)~~assembly components, including shroud plates and ~~(b)~~ former plates	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	Chapter XI.M16A, "PWR Vessel Internals" ~~Primary components (identified in the "Structure and Components" column) no Expansion components~~	No
IV.B3.RP-316	IV.B3-9 (R-162)	Core shroud assemblies (for bolted core shroud assemblies): barrel-shroud bolts ~~with neutron exposures greater than 3 dpa~~	Stainless steel	Reactor coolant and neutron flux	Cracking due to irradiation-assisted stress~~-~~ corrosion cracking or fatigue	~~Chapter XI.M2, "Water Chemistry," and~~ Chapter XI.M16A, ~~"~~"PWR Vessel Internals~~"~~ ~~Expansion components (identified in the "Structure and Components" column)~~ " and Chapter XI.M2, "Water Chemistry" (for ~~Primary components see AMR Item IV.B3.RP-314)~~SCC mechanisms only)	No
IV.B3.RP-317	IV.B3-7 (R-165)	Core shroud assemblies (for bolted core shroud assemblies): barrel-shroud bolts ~~with neutron exposures greater than 3 dpa~~	Stainless steel~~; nickel alloy~~	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	Chapter XI.M16A, "PWR Vessel Internals" ~~Expansion components (identified in the "Structure and Components" column) (for Primary components see AMR Item IV.B3.RP-315)~~	No

IV B3	REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM Reactor Vessel Internals (PWR) - Combustion Engineering						
Item	**Link**	**Structure and/or Component**	**Material**	**Environment**	**Aging Effect/ Mechanism**	**Aging Management Program (AMP)**	**Further Evaluation**
IV.B3.RP-314	IV.B3-9 (R-162)	Core shroud assemblies (for bolted core shroud assemblies): core shroud bolts ~~(accessible)~~	Stainless steel	Reactor coolant and neutron flux	Cracking due to irradiation-assisted stress~~-~~ corrosion cracking ~~and~~or fatigue	~~Chapter XI.M2, "Water Chemistry," and~~ Chapter XI.M16A, "PWR Vessel Internals" ~~Primary components (identified in the "Structure and Components" column)~~ and Chapter XI.M2, "Water Chemistry" (for ~~Expansion components see AMR Items IV.B3.RP-316, IV.B3.RP-330, and IV.B3.RP-358)~~SCC mechanisms only)	No
IV.B3.RP-315	IV.B3-7(R-165)	Core shroud assemblies (for bolted core shroud assemblies): core shroud bolts ~~(accessible)~~	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 dimensions due to void swelling or distortion	Chapter XI.M16A, "PWR Vessel Internals~~,"~~ ~~Primary components (identified in the "Structure and Components" column) (for Expansion components see AMR Items IV.B3.RP-317, and IV.B3.RP-331)~~"	No
IV.B3.RP-359		Core shroud ~~assemblies (welded): (~~assembly (designs assembled in two vertical sections): core shroud ~~plates and (b)~~ plate-to-former ~~plates~~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	Chapter XI.M16A, "PWR Vessel Internals~~,"~~ ~~Primary components (identified in the "Structure and Components" column) no Expansion components~~"	No

IV B3	REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM Reactor Vessel Internals (PWR) - Combustion Engineering						
Item	**Link**	**Structure and/or Component**	**Material**	**Environment**	**Aging Effect/ Mechanism**	**Aging Management Program (AMP)**	**Further Evaluation**
IV.B3.RP-322		Core shroud assembly (~~for welded core shrouds~~designs assembled in two vertical sections): ~~Core shroud plate-former plate weld (a) The axial and horizontal weld seams at the core shroud re-entrant corners as visible from the core side of the shroud, within six inches of the central flange and horizontal stiffeners, and (b) the horizontal stiffeners in~~ core shroud plate-to-former plate ~~weld~~welds	Stainless steel	Reactor coolant and neutron flux	Cracking due to irradiation-assisted stress~~-~~ corrosion cracking	~~Chapter XI.M2, "Water Chemistry," and~~ Chapter XI.M16A, "PWR Vessel Internals" ~~Primary components (identified in the "Structure and Components" column) (for Expansion components see AMR Item IV.B3.RP-323)~~and Chapter XI.M2, "Water Chemistry"	No
IV.B3.RP-326		Core shroud assembly (~~for welded core shrouds~~designs assembled in two vertical sections): ~~gap between~~assembly components, including monitoring of the ~~upper and lower plates~~gap opening at the core shroud re-entrant corners	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	Chapter XI.M16A, "PWR Vessel Internals" ~~Primary components (identified in the "Structure and Components" column) no Expansion components~~	No

IV B3	REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM Reactor Vessel Internals (PWR) - Combustion Engineering						
Item	**Link**	**Structure and/or Component**	**Material**	**Environment**	**Aging Effect/ Mechanism**	**Aging Management Program (AMP)**	**Further Evaluation**
IV.B3.RP-~~323~~326a		Core shroud assembly (~~for welded core shrouds~~designs assembled in two vertical sections): ~~remaining axial welds in~~ assembly components, including monitoring of the gap opening at the core shroud ~~plate-to-former plate~~re-entrant corners	Stainless steel	Reactor coolant and neutron flux	Cracking due to ~~irradiation-assisted~~ stress~~-~~corrosion cracking or fatigue	~~Chapter XI.M2, "Water Chemistry," and~~ Chapter XI.M16A, "PWR Vessel Internals" ~~Expansion components (identified in the "Structure and Components" column)~~ and Chapter XI.M2, "Water Chemistry" (for ~~Primary components see AMR Item IV.B3.RP-322)~~SCC mechanisms only)	No
IV.B3.RP-~~324~~323		~~Core shroud assembly (for welded core shrouds with full-height shroud plates): axial weld seams at the core shroud re-entrant corners, at the core mid-plane (+3 feet in height) as visible from the core side of the shroud~~ Core shroud assembly (designs assembled in two vertical sections): remaining axial welds	Stainless steel	Reactor coolant and neutron flux	Cracking due to irradiation-assisted stress~~-~~corrosion cracking	~~Chapter XI.M2, "Water Chemistry," and~~ Chapter XI.M16A, "PWR Vessel Internals" ~~Primary components (identified in the "Structure and Components" column) (for Expansion components see AMR Item IV.B3.RP-325)~~and Chapter XI.M2, "Water Chemistry"	No

IV B3	REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM Reactor Vessel Internals (PWR) - Combustion Engineering						
Item	**Link**	**Structure and/or Component**	**Material**	**Environment**	**Aging Effect/ Mechanism**	**Aging Management Program (AMP)**	**Further Evaluation**
IV.B3.RP-~~360~~359a		~~Core shroud assembly (for welded core shrouds with full-height shroud plates): axial weld seams at the core shroud re-entrant corners, at the core mid-plane (+3 feet in height) as visible from the core side of the shroud~~ Core shroud assembly (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	Chapter XI.M16A, ~~"~~"PWR Vessel Internals~~" Primary components (identified in the "Structure and Components" column) (for Expansion components see AMR Item IV.B3.RP-361)~~"	No
IV.B3.RP-~~325~~324		Core shroud assembly (~~for welded core shrouds~~ designs assembled with full-height shroud plates): ~~remaining~~shroud plate axial ~~welds, ribs, and rings~~weld seams at the core shroud re-entrant corners	Stainless steel	Reactor coolant and neutron flux	Cracking due to irradiation-assisted stress~~-~~ corrosion cracking	~~Chapter XI.M2, "Water Chemistry," and~~ Chapter XI.M16A, "PWR Vessel Internals" and Chapter XI.M2, "Water Chemistry" ~~Expansion components (identified in the "Structure and Components" column) (for Primary components see AMR Item IV.B3.RP-324)~~	No

IV B3	REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM Reactor Vessel Internals (PWR) - Combustion Engineering						
Item	**Link**	**Structure and/or Component**	**Material**	**Environment**	**Aging Effect/ Mechanism**	**Aging Management Program (AMP)**	**Further Evaluation**
IV.B3.RP-~~361~~360		Core shroud assembly (~~for welded core shrouds~~ designs assembled with full-height shroud plates): ~~remaining~~shroud plate axial ~~welds, ribs, and rings~~weld seams at the core shroud re-entrant corners	Stainless steel	Reactor coolant and neutron flux	Loss of fracture toughness due to neutron irradiation embrittlement	Chapter XI.M16A, "PWR Vessel Internals" ~~Expansion components (identified in the "Structure and Components" column) (for Primary components see AMR Item IV.B3.RP-360)~~	No
IV.B3.RP-~~362~~325		Core ~~support barrel~~shroud assembly~~: lower cylinder~~ (designs assembled with full-height shroud plates): remaining axial welds, ribs, and rings	Stainless steel	Reactor coolant and neutron flux	~~Loss of fracture toughness~~Cracking due to ~~neutron~~ irradiation ~~embrittlement~~ assisted stress corrosion cracking	Chapter XI.M16A, ~~"~~"PWR Vessel Internals~~"~~ ~~Expansion components (identified in the "Structure~~" and ~~Components" column)~~Chapter XI.M2, "Water Chemistry"" ~~(for Primary components see AMR Item IV.B3.RP-327)~~	No
IV.B3.RP-~~329~~361	~~IV.B3-15(R-155)~~	Core ~~support barrel~~shroud assembly~~: lower cylinder~~ (designs assembled with full-height shroud plates): remaining axial welds, ribs, and ~~remaining core barrel assembly welds~~rings	Stainless steel	Reactor coolant and neutron flux	~~Cracking~~Loss of fracture toughness due to ~~stress corrosion cracking~~neutron irradiation embrittlement	~~Chapter XI.M2, "Water Chemistry," and~~ Chapter XI.M16A, ~~"~~"PWR Vessel Internals~~"~~" ~~Expansion components (identified in the "Structure and Components" column) (for Primary components see AMR Item IV.B3.RP-327)~~	No
IV.B3.RP-~~333~~362		Core support barrel assembly: lower ~~flange weld, if fatigue life cannot be demonstrated by TLAA~~cylinder circumferential (girth) welds	Stainless steel	Reactor coolant and neutron flux	~~Cracking~~Loss of fracture toughness due to ~~fatigue~~neutron irradiation embrittlement	~~Chapter XI.M2, "Water Chemistry," and~~ Chapter XI.M16A, ~~"~~"PWR Vessel Internals~~"~~" ~~Primary components (identified in the "Structure and Components" column) no Expansion components~~	~~Yes, evaluate to determine the potential locations and extent of fatigue cracking~~No

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

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
IV.B3.RP-~~389~~ 362a		Core support barrel assembly: lower ~~flange weld (if fatigue analysis exists)~~cylinder circumferential (girth) welds	Stainless steel	Reactor coolant and neutron flux	~~Cumulative fatigue damage due to fatigue~~Cracking due to stress corrosion cracking or irradiation-assisted stress corrosion cracking	Chapter XI.M16A, "PWR Vessel Internals" and Chapter XI.M2, "Water Chemistry"	~~Yes, TLAA~~No
IV.B3.RP-~~328~~362b	~~IV.B3-15(R-155)~~	Core support barrel assembly: ~~surfaces of the~~ lower ~~core barrel flange weld (accessible surfaces)~~cylinder vertical (axial) welds	Stainless steel	Reactor coolant and neutron flux	~~Cracking~~Loss of fracture toughness due to ~~stress corrosion cracking and fatigue~~neutron irradiation embrittlement	~~Chapter XI.M2, "Water Chemistry," and~~ Chapter XI.M16A, ~~-~~"PWR Vessel Internals~~" Primary components (identified in the "Structure and Components" column) no Expansion components~~"	No
IV.B3.RP-~~332~~362c	~~IV.B3-17(R-156)~~	Core support barrel assembly: ~~upper core barrel flange~~lower cylinder vertical (axial) welds	Stainless steel	Reactor coolant and neutron flux	~~Loss of material due to wear~~Cracking due to stress corrosion cracking or irradiation-assisted stress corrosion cracking	Chapter XI.M16A, "PWR Vessel Internals" ~~Existing Program components (identified in the "Structure~~ and ~~Components" column) no Expansion components~~Chapter XI.M2, "Water Chemistry"	No
IV.B3.RP-~~327~~329	IV.B3-15(R-155)	Core support barrel assembly: upper cylinder (base metal and welds) and upper core ~~support~~ barrel flange ~~weld (accessible surfaces)~~(flange base metal)	Stainless steel	Reactor coolant and neutron flux	Cracking due to stress~~-~~ corrosion cracking	~~Chapter XI.M2, "Water Chemistry," and~~ Chapter XI.M16A, "PWR Vessel Internals" ~~Primary components (identified in the "Structure and Components" column) (for Expansion components see AMR Items IV.B3.RP-329, IV.B3.RP-335, IV.B3.RP-362, IV.B3.RP-363, IV.B3.RP-364)~~and Chapter XI.M2, "Water Chemistry"	No
IV.B3.RP-~~357~~333		~~Incore instrumentation (ICI): ICI thimble tubes - lower~~Core support barrel assembly: lower flange	~~Zircaloy-~~4Stainless steel	Reactor coolant and neutron flux	~~Loss of material~~Cracking due to ~~wear~~stress corrosion cracking or fatigue	~~A plant-specific aging management program is to be evaluated~~Chapter XI.M16A, "PWR Vessel Internals" and Chapter XI.M2, "Water Chemistry" (for SCC mechanisms only)	~~Yes, plant-specific~~No

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B3 Reactor Vessel Internals (PWR) - Combustion Engineering

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
IV.B3.RP-~~336~~328	IV.B3-~~22~~15(R-~~170)~~155)	~~Lower support structure: A286 fuel alignment pins (all plants with core shroud assembled in two vertical sections)~~Core support barrel assembly: lower core barrel flange weld	Stainless steel	Reactor coolant and neutron flux	Cracking~~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~~corrosion cracking or fatigue	Chapter XI.M16A, "PWR Vessel Internals" ~~Existing Program components (identified in the "Structure~~ and ~~Components" column) no Expansion components~~Chapter XI.M2, "Water Chemistry" (for SCC mechanisms only)	No
IV.B3.RP-~~334~~332	IV.B3-~~23~~17(R-~~167)~~156)	~~Lower support structure: A286 fuel alignment pins (all plants with core shroud assembled with full-height shroud plates)~~Core support barrel assembly: upper core barrel flange	Stainless steel	Reactor coolant and neutron flux	~~Cracking~~Loss of material due to ~~irradiation-assisted stress corrosion cracking and fatigue~~wear	~~'Chapter XI.M2, "Water Chemistry," and~~ Chapter XI.M16A, "PWR Vessel Internals" ~~Existing Program components (identified in the "Structure and Components" column) no Expansion components~~	No
IV.B3.RP-~~364~~327	IV.B3-15(R-155)	~~Lower~~Core support ~~structure:~~barrel assembly: upper core ~~support column~~barrel flange weld	~~Cast austenitic stainless~~Stainless steel	Reactor coolant and neutron flux	~~Loss of fracture toughness~~Cracking due to ~~neutron irradiation and thermal embrittlement~~stress corrosion cracking	Chapter XI.M16A, ~~"~~"PWR Vessel Internals~~" Expansion components (identified in the "Structure~~" and ~~Components" column) (for Primary components see AMR Item IV.B3RP-327)~~Chapter XI.M2, "Water Chemistry"	No
IV.B3.RP-~~363~~357		~~Lower support structure: core support column~~Incore instruments (ICI): ICI thimble tubes - lower	~~Stainless steel~~Zircaloy-4	Reactor coolant and neutron flux	Loss of ~~fracture toughness~~material due to ~~neutron irradiation embrittlement~~wear	Chapter XI.M16A, ~~"~~"PWR Vessel Internals~~" Expansion components (identified in the "Structure and Components" column) (for Primary components see AMR Item IV.B3RP-327)~~"	No

IV B3	REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM Reactor Vessel Internals (PWR) - Combustion Engineering						
Item	**Link**	**Structure and/or Component**	**Material**	**Environment**	**Aging Effect/ Mechanism**	**Aging Management Program (AMP)**	**Further Evaluation**
IV.B3.RP-~~330~~336	IV.B3-~~23~~22(R-~~167)~~170)	Lower support structure~~: core support column bolts~~ (designs assembled in two vertical sections): fuel alignment pins	Stainless steel	Reactor coolant and neutron flux	Loss of material~~Cracking~~ due to wear; loss of fracture toughness due to neutron irradiation~~-assisted~~ embrittlement; loss of preload due to thermal and irradiation enhanced stress ~~corrosion cracking and fatigue~~relaxation or creep	~~Chapter XI.M2, "Water Chemistry," and~~ Chapter XI.M16A, "PWR Vessel Internals" ~~Expansion components (identified in the "Structure and Components" column) (for Primary components see AMR Item 'IV.B3.RP-314)~~	No
IV.B3.RP-~~331~~334	IV.B3-23(R-167)	Lower support structure~~: core support column bolts~~ (designs assembled in two vertical sections or with full-height shroud plates): fuel alignment pins	Stainless steel	Reactor coolant and neutron flux	~~Loss of fracture toughness~~Cracking due to ~~neutron~~stress corrosion cracking, irradiation ~~embrittlement~~ assisted stress corrosion cracking, or fatigue	Chapter XI.M16A, "PWR Vessel Internals" ~~Expansion components (identified in the "Structure~~ and ~~Components" column)~~ Chapter XI.M2, "Water Chemistry" (for ~~Primary components see AMR Item 'IV.B3.RP-315)~~SCC mechanisms only)	No
IV.B3.RP-~~335~~334a	IV.B3-~~23~~22(R-~~167)~~170)	Lower support structure~~: core support column welds, applicable to all plants except those~~ (designs assembled in two vertical sections or with full-height shroud plates): fuel alignment pins	Stainless steel	Reactor coolant and neutron flux	~~Cracking~~Loss of material due to ~~stress corrosion cracking,~~wear; loss of fracture toughness due to neutron irradiation~~-assisted stress corrosion cracking,~~ embrittlement; loss of preload due to thermal and ~~fatigue~~irradiation enhanced stress relaxation or creep	~~Chapter XI.M2, "Water Chemistry," and~~ Chapter XI.M16A, "PWR Vessel Internals" ~~Expansion components (identified in the "Structure and Components" column) (for Primary components see AMR Item IV.B3.RP-327)~~	No

IV B3	REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM Reactor Vessel Internals (PWR) - Combustion Engineering						
Item	**Link**	**Structure and/or Component**	**Material**	**Environment**	**Aging Effect/ Mechanism**	**Aging Management Program (AMP)**	**Further Evaluation**
IV.B3.RP-~~365~~364		Lower support structure~~:~~ (all plants): core support ~~plate~~column welds	Stainless steel (including CASS)	Reactor coolant and neutron flux	Loss of fracture toughness due to neutron irradiation embrittlement and for column welds made from CASS, thermal embrittlement	Chapter XI.M16A, "PWR Vessel Internals" ~~Primary component (identified in the "Structure and Components" column) no Expansion components~~	No
IV.B3.RP-~~343~~363		Lower support structure~~:~~ (all plants): core support ~~plate (applicable to plants with a core support plate), if fatigue life cannot be demonstrated by TLAA~~column welds	Stainless steel	Reactor coolant and neutron flux	Cracking due to stress corrosion cracking, irradiation-assisted stress corrosion cracking, or fatigue	~~Chapter XI.M2, "Water Chemistry", and~~ Chapter XI.M16A, "PWR Vessel Internals" and Chapter XI.M2, "Water Chemistry" (for SCC mechanisms only) ~~Primary components (identified in the "Structure and Components" column) no Expansion components~~	~~Yes, evaluate to determine the potential locations and extent of fatigue cracking~~No
IV.B3.RP-~~390~~330	IV.B3-23(R-167)	Lower support structure: core support ~~plate (applicable to plants with a core support plate), if fatigue analysis exists~~column bolts	Stainless steel	Reactor coolant and neutron flux	~~Cumulative~~Cracking due to irradiation-assisted stress corrosion cracking or fatigue ~~damage due to fatigue~~	~~Fatigue is a time-limited aging analysis (TLAA) to be evaluated for the period of extended operation. See the SRP, Section 4.3 "Metal Fatigue," for acceptable methods for meeting the requirements of 10 CFR 54.21(c)(1).~~Chapter XI.M16A, "PWR Vessel Internals" and Chapter XI.M2, "Water Chemistry" (for SCC mechanisms only)	~~Yes, TLAA~~No
IV.B3.RP-~~342~~331		Lower support structure: ~~deep beams (applicable assemblies with full height shroud plates)~~core support column bolts	Stainless steel	Reactor coolant and neutron flux	~~Cracking~~Loss of fracture toughness due to ~~stress corrosion cracking,~~neutron irradiation~~-assisted stress corrosion cracking, and fatigue~~ embrittlement	~~Chapter XI.M2, "Water Chemistry," and~~ Chapter XI.M16A, "PWR Vessel Internals" ~~Primary components (identified in the "Structure and Components" column) no Expansion components~~	No

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B3 Reactor Vessel Internals (PWR) - Combustion Engineering

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
IV.B3.RP-~~366~~335	IV.B3-23(R-167)	Lower support structure~~: deep beams (applicable assemblies~~ (designs except those assembled with full~~-~~-height shroud plates~~)~~): lower core support beams	Stainless steel	Reactor coolant and neutron flux	~~Loss of fracture toughness~~Cracking due to ~~neutron irradiation embrittlement~~stress corrosion cracking or fatigue	Chapter XI.M16A, “PWR Vessel Internals” ~~Primary components (identified in the "Structure~~ and ~~Components" column) no Expansion components~~Chapter XI.M2, “Water Chemistry” (for SCC mechanisms only)	No
IV.B3.RP-365		Lower support structure (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	Chapter XI.M16A, "PWR Vessel Internals" ~~Primary component (identified in the "Structure and Components" column) no Expansion components~~	No
IV.B3.RP-~~24~~343	~~IV.B3-25(RP-24)~~	~~Reactor vessel internal components~~Lower support structure (designs with a core support plate): core support plate	Stainless steel~~; nickel alloy~~	Reactor coolant and neutron flux	~~Loss of material~~Cracking due to ~~pitting and crevice corrosion~~fatigue	Chapter XI.~~M2, “Water Chemistry”~~M16A, “PWR Vessel Internals”	No
IV.B3.RP-~~309~~342		~~Reactor vessel internal components (inaccessible locations)~~Lower support structure (designs with core shrouds assembled with full height shroud plates): deep beams	Stainless steel~~; nickel alloy~~	Reactor coolant and neutron flux	Cracking due to stress~~-~~-corrosion cracking, ~~and~~ irradiation-assisted stress~~-~~-corrosion cracking, or fatigue	~~Chapter XI.M2, “Water Chemistry,” and~~ Chapter XI.M16A, “PWR Vessel Internals” and Chapter XI.M2, “Water Chemistry” (for SCC mechanisms only)	~~Yes, if accessible Primary, Expansion or Existing program components indicate aging effects that need management~~No

IV B3	REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM Reactor Vessel Internals (PWR) - Combustion Engineering						
Item	**Link**	**Structure and/or Component**	**Material**	**Environment**	**Aging Effect/ Mechanism**	**Aging Management Program (AMP)**	**Further Evaluation**
IV.B3.RP-~~311~~366		~~Reactor vessel internal components (inaccessible locations)~~Lower support structure (designs with core shrouds assembled with full height shroud plates): deep beams	Stainless steel~~; nickel alloy~~	Reactor coolant and neutron flux	Loss of fracture toughness due to neutron irradiation embrittlement~~; change in dimension due to void swelling; loss of preload due to thermal and irradiation enhanced stress relaxation; loss of material due to wear~~	Chapter XI.M16A, "PWR Vessel Internals"	~~Yes, if accessible Primary, Expansion or Existing program components indicate aging effects that need management~~No
IV.B3.RP-339	IV.B3-24(R-53)	Reactor vessel internal components	Stainless steel; nickel alloy	Reactor coolant and neutron flux	Cumulative fatigue damage due to fatigue	Fatigue is a time-limited aging analysis (TLAA) to be evaluated for the period of extended operation. See the SRP, Section 4.3 "Metal Fatigue," for acceptable methods for meeting the requirements of 10 CFR 54.21(c)(1).	Yes, TLAA
IV.B3.RP-306		Reactor internal "No Additional Measures" components~~Reactor vessel internal components with no additional measures~~	Stainless steel; nickel alloy	Reactor coolant and neutron flux	No additional aging management for reactor internal "No Additional Measures" components unless required by ASME Section XI, Examination Category B-N-3 or relevant operating experience exists~~Cracking due to stress corrosion cracking, and irradiation-assisted stress corrosion cracking~~	~~Chapter XI.M2, "Water Chemistry," and~~ Chapter XI.M16A, "PWR Vessel Internals" ~~Note: Components with no additional measures are not uniquely identified in GALL tables. Components with no additional measures are defined in Section 3.3.1 of MRP-227, "Materials Reliability Program: Pressurized Water Reactor Internals Inspection and Evaluation Guidelines"~~	No

IV B3	REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM Reactor Vessel Internals (PWR) - Combustion Engineering						
Item	**Link**	**Structure and/or Component**	**Material**	**Environment**	**Aging Effect/ Mechanism**	**Aging Management Program (AMP)**	**Further Evaluation**
IV.B3.RP-~~307~~24	IV.B3-25(RP-24)	Reactor vessel internal components ~~with no additional measures~~	Stainless steel; nickel alloy	Reactor coolant and neutron flux	Loss of ~~fracture toughness~~material due to ~~neutron irradiation embrittlement; change in dimension due to void swelling; loss of preload due to thermal and irradiation enhanced stress relaxation; loss of material due to wear~~pitting and crevice corrosion	Chapter XI.~~M16A, "PWR Vessel Internals" Note: Components with no additional measures are not uniquely identified in GALL tables - Components with no additional measures are defined in Section 3.3.1 of MRP-227, "Materials Reliability Program: Pressurized~~ M2, "Water ~~Reactor Internals Inspection and Evaluation Guidelines"~~Chemistry"	No
IV.B3.RP-382	IV.B3-22(R-170)	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, stress corrosion cracking, or irradiation-assisted stress corrosion cracking; ~~Loss~~loss of material due to wear	Chapter XI.M1, "ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD" or Chapter XI.M16A, "PWR Vessel Internals," by invoking applicable 10 CFR 50.55a and ASME Section XI inservice inspection requirements	No
IV.B3.RP-338		Upper internals assembly~~: fuel alignment plate (applicable to plants~~ (designs with core shrouds assembled with full height shroud plates~~), if fatigue life cannot be demonstrated by TLAA~~): fuel alignment plate	Stainless steel	Reactor coolant and neutron flux	Cracking due to fatigue	~~'Chapter XI.M2, "Water Chemistry," and~~ Chapter XI.M16A, "PWR Vessel Internals" ~~Primary components (identified in the "Structure and Components" column) no Expansion components~~	~~Yes, evaluate to determine the potential locations and extent of fatigue cracking~~No

IV B3	REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM Reactor Vessel Internals (PWR) - Combustion Engineering						
Item	**Link**	**Structure and/or Component**	**Material**	**Environment**	**Aging Effect/ Mechanism**	**Aging Management Program (AMP)**	**Further Evaluation**
IV.B3.RP-~~391~~400		~~Upper internals assembly: fuel alignment plate (applicable to plants with core shrouds assembled with full height shroud plates), if fatigue analysis exists~~ Core Support Barrel Assembly: thermal shield positioning pins	Stainless steel	Reactor coolant and neutron flux	~~Cumulative fatigue damage due to fatigue~~Cracking due to stress corrosion cracking, irradiation-assisted stress corrosion cracking or fatigue; loss of material due to wear	~~Fatigue is a time-limited aging analysis (TLAA) to be evaluated for the period of extended operation. See the SRP, Section 4.3 "Metal Fatigue," for acceptable methods for meeting the requirements of 10 CFR 54.21(c)(1).~~Chapter XI.M16A, "PWR Vessel Internals" and Chapter XI.M2, "Water Chemistry" (for SCC mechanisms only)	~~Yes, TLAA~~No

(4) Mark-up of changes to GALL Report Chapter IV.B4

B4. REACTOR VESSEL INTERNALS (PWR) - BABCOCK AND WILCOX

Systems, Structures, and Components

This section addresses the Babcock and Wilcox (B&W) pressurized-water reactor (PWR) vessel internals and consists, which consist of components in the plenum cover and plenum cylinderassembly, the upper grid assembly, the control rod guide tube (CRGT) assembly, the core support shield assembly, the core barrel assembly, the lower grid assembly, and the flow distributor assembly. Based on Regulatory Guide 1.26, "Quality Group Classifications and Standards for Water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants," all structures and components that comprise the reactor vessel are governed by Group A or B Quality Standards.incore monitoring instrumentation (IMI) guide tube assembly, and the flow distributor assembly.

Common miscellaneous material/environment combinations where aging effects are not expected to degrade the ability of the structure or component to perform its intended function for the period of extended operation are included in IV.E.

System Interfaces

The systems that interface with the reactor vessel internals include the reactor pressure vessel (IV.A2).

Inspection Plan

An applicant will submit an inspection plan for reactor internals to the NRC for review and approval with the application for license renewal in accordance with Chapter XI.M16A, "PWR Vessel Internals."

IV B4	REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM Reactor Vessel Internals (PWR) – Babcock & Wilcox						
Item	**Link**	**Structure and/or Component**	**Material**	**Environment**	**Aging Effect/ Mechanism**	**Aging Management Program (AMP)**	**Further Evaluation**
IV.B4.RP-242	IV.B4-4 (R-183)	Control rod guide tube (CRGT) assembly: ~~accessible surfaces at four screw locations (every 90 degrees) for~~ CRGT spacer castings	Cast austenitic stainless steel	Reactor coolant and neutron flux	Loss of fracture toughness due to thermal aging embrittlement	Chapter XI.M16A, "PWR Vessel Internals" ~~Expansion components (identified in the "Structure and Components" column) (for Primary components see AMR Items IV.B4.RP-253 and IV.B4.RP-258)~~	No
IV.B4.RP-242a		Control rod guide tube (CRGT) assembly: CRGT spacer castings	Stainless steel (including CASS)	Reactor coolant and neutron flux	Cracking due to stress corrosion cracking or fatigue	Chapter XI.M16A, "PWR Vessel Internals" and Chapter XI.M2, "Water Chemistry" (for SCC mechanisms only)	No
IV.B4.RP-245	IV.B4-13 (R-194)	Core barrel assembly~~: (a) upper thermal shield bolts; (b)~~ (applicable to Crystal River Unit 3 or Davis Besse only): surveillance specimen holder tube ~~bolts (Davis-Besse, only); (c) surveillance specimen tube holder~~(SSHT) studs~~,~~ ~~and~~ /nuts ~~(Crystal River Unit 3, only)~~or bolts	~~Stainless steel; nickel~~Nickel alloy	Reactor coolant and neutron flux	Cracking due to stress~~-~~ corrosion cracking	~~Chapter XI.M2, "Water Chemistry," and~~ Chapter XI.M16A, "PWR Vessel Internals" and Chapter XI.M2, "Water Chemistry" ~~Expansion components (identified in the "Structure and Components" column) (for Primary components see AMR Items IV.B4.RP-247 and IV.B4.RP-248)~~	No
IV.B4.RP-245a		Core barrel assembly (applicable to Crystal River Unit 3 or Davis Besse only): surveillance specimen holder tube (SSHT) stud or bolt locking devices	Nickel alloy	Reactor coolant and neutron flux	Cracking due to fatigue	Chapter XI.M16A, "PWR Vessel Internals"	No

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

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
IV.B4.RP-245b		Core barrel assembly (applicable to CR-3 or DB only): surveillance specimen holder tube (SSHT) stud or bolt locking devices	Nickel alloy	Reactor coolant and neutron flux	Loss of material due to wear; changes in dimensions due to void swelling or distortion	Chapter XI.M16A, "PWR Vessel Internals"	No
IV.B4.RP-247	IV.B4-13 (R-194)	Core barrel assembly: ~~accessible~~ lower core barrel (LCB) bolts ~~and locking devices~~	Stainless steel; nickel alloy	Reactor coolant and neutron flux	Cracking due to stress~~-~~ corrosion cracking	~~Chapter XI.M2, "Water Chemistry," and~~ Chapter XI.M16A, "PWR Vessel Internals" and Chapter XI.M2, "Water Chemistry" ~~Primary components (identified in the "Structure and Components" column) (for Expansion components see AMR Items IV.B4.RP-245, IV.B4.RP-246, IV.B4.RP-254, and IV.B4.RP-256)~~	No
IV.B4.RP-247a		Core barrel assembly: lower core barrel (LCB) bolt locking devices	Stainless steel; nickel alloy	Reactor coolant and neutron flux	Cracking due to fatigue	Chapter XI.M16A, "PWR Vessel Internals"	No
IV.B4.RP-247b		Core barrel assembly: lower core barrel (LCB) bolt locking devices	Stainless steel; nickel alloy	Reactor coolant and neutron flux	Loss of material due to wear; changes in dimensions due to void swelling or distortion	Chapter XI.M16A, "PWR Vessel Internals"	No
IV.B4.RP-249	IV.B4-12 (R-196)	Core barrel assembly: baffle ~~plate accessible surfaces within one inch around each baffle plate flow and bolt hole~~plates	Stainless steel	Reactor coolant and neutron flux	Loss of fracture toughness due to neutron irradiation embrittlement	Chapter XI.M16A, "PWR Vessel Internals" ~~Primary components (identified in the "Structure and Components" column) (for Expansion components see AMR Item IV.B4.RP-250)~~	No
IV.B4.RP-249a		Core barrel assembly: baffle plates	Stainless steel	Reactor coolant and neutron flux	Cracking due to irradiation-assisted stress corrosion cracking	Chapter XI.M16A, "PWR Vessel Internals" and Chapter XI.M2, "Water Chemistry"	No

IV B4	REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM Reactor Vessel Internals (PWR) – Babcock & Wilcox						
Item	**Link**	**Structure and/or Component**	**Material**	**Environment**	**Aging Effect/ Mechanism**	**Aging Management Program (AMP)**	**Further Evaluation**
IV.B4.RP-241	IV.B4-7 (R-125)	Core barrel assembly: baffle-to-former bolts and screws	Stainless steel	Reactor coolant and neutron flux	Cracking due to stress corrosion cracking, irradiation-assisted stress corrosion cracking, fatigue, and overload	Chapter XI.M16A, "PWR Vessel Internals" and Chapter XI.M2, "Water Chemistry" (for SCC mechanisms only)	No
IV.B4.RP-~~241~~241a	~~IV.B4-7(R-125)~~	Core barrel assembly: ~~baffle/former assembly: (a) accessible baffle-to-former bolts and screws; (b) accessible~~ 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 stress~~-~~-corrosion cracking, irradiation-assisted stress~~-~~-corrosion cracking, fatigue, and overload	~~Chapter XI.M2, "Water Chemistry," and~~ Chapter XI.M16A, "PWR Vessel Internals" ~~Primary Components (identified in the "Structure and Components" column)~~ and Chapter XI.M2, "Water Chemistry" (for ~~Expansion components see AMR Items IV.B4.RP-244 and IV.B4.RP-375)~~SCC mechanisms only)	No
IV.B4.RP-240	IV.B4-1 (R-128)	Core barrel assembly: baffle~~/former assembly: (a) accessible baffle~~-to-former bolts and screws~~; (b) accessible locking devices (including welds) of baffle-to-former bolts~~	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	Chapter XI.M16A, "PWR Vessel Internals~~.~~"" ~~Primary components (identified in the "Structure and Components" column) (for Expansion components see AMR Item IV.B4.RP-243.)~~	No
IV.B4.RP-240a		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	Chapter XI.M16A, "PWR Vessel Internals"	No

IV B4	REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM Reactor Vessel Internals (PWR) – Babcock & Wilcox						
Item	**Link**	**Structure and/or Component**	**Material**	**Environment**	**Aging Effect/ Mechanism**	**Aging Management Program (AMP)**	**Further Evaluation**
IV.B4.RP-250	IV.B4-12 (R-196)	Core barrel assembly: core barrel cylinder (including vertical and circumferential seam welds); former plates	Stainless steel	Reactor coolant and neutron flux	Loss of fracture toughness due to neutron irradiation embrittlement	Chapter XI.M16A, "PWR Vessel Internals" ~~Expansion components (identified in the "Structure and Components" column) (for Primary components see AMR Item IV.B4.RP-249)~~	No
IV.B4.RP-250a		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 stress corrosion cracking or fatigue	Chapter XI.M16A, "PWR Vessel Internals" and Chapter XI.M2, "Water Chemistry" (for SCC mechanisms only)	No
IV.B4.RP-375		Core barrel assembly: internal baffle-to-baffle bolts	Stainless steel	Reactor coolant and neutron flux	Cracking due to irradiation-assisted stress corrosion cracking, fatigue, or overload	~~Chapter XI.M2, "Water Chemistry," and~~ Chapter XI.M16A, "PWR Vessel Internals" ~~Expansion components (identified in the "Structure and Components" column)~~ and Chapter XI.M2, "Water Chemistry" (for ~~Primary components see AMR Item IV.B4.RP-241~~SCC mechanisms only)	No
IV.B4.RP-375a		Core barrel assembly: internal baffle-to-baffle 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	Chapter XI.M16A, "PWR Vessel Internals"	No
IV.B4.RP-244	IV.B4-7 (R-125)	Core barrel assembly; external baffle-to-baffle bolts-and core barrel-to-former bolts;	Stainless steel	Reactor coolant and neutron flux	Cracking due to irradiation-assisted stress corrosion cracking, fatigue, and overload	Chapter XI.M16A, "PWR Vessel Internals" and Chapter XI.M2, "Water Chemistry" (for SCC mechanisms only)	No

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

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
IV.B4.RP-~~244~~244a	~~IV.B4-7(R-125)~~	Core barrel assembly~~; (a) external baffle-to-baffle bolts; (b) core barrel-to-former bolts; (c)~~: 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 stress~~-~~ corrosion cracking, or fatigue	~~Chapter XI.M2, "Water Chemistry," and~~ Chapter XI.M16A, ~~"~~"PWR Vessel Internals~~"~~ ~~Expansion components (identified in the "Structure and Components" column)~~ " and Chapter XI.M2, "Water Chemistry" (for ~~Primary components see AMR Item IV.B4.RP-241)~~SCC mechanisms only)	No
IV.B4.RP-243	IV.B4-1 (R-128)	Core barrel assembly~~; (a) external baffle-to-baffle bolts; (b) core barrel-to-former bolts; (c) locking devices (including welds) of~~: external baffle-to-baffle bolts and core barrel-to-former bolts~~; (d) internal baffle-to-baffle 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	Chapter XI.M16A, "PWR Vessel Internals" ~~Expansion components (identified in the "Structure and Components" column) (for Primary components see AMR Item IV.B4.RP-240)~~	No
IV.B4.RP-243a		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	Loss of fracture toughness due to neutron irradiation embrittlement; loss of material due to wear	Chapter XI.M16A, "PWR Vessel Internals"	No
IV.B4.RP-248	IV.B4-12 (R-196)	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 stress~~-~~ corrosion cracking	~~Chapter XI.M2, "Water Chemistry," and~~ Chapter XI.M16A, "PWR Vessel Internals" and Chapter XI.M2, "Water Chemistry" ~~Primary components (identified in the "Structure and Components" column) (for Expansion components see AMR Items IV.B4.RP-245, IV.B4.RP-246, IV.B4.RP-254, IV.B4.RP-247, and IV.B4.RP-256)~~	No

IV B4	REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM Reactor Vessel Internals (PWR) – Babcock & Wilcox						
Item	**Link**	**Structure and/or Component**	**Material**	**Environment**	**Aging Effect/ Mechanism**	**Aging Management Program (AMP)**	**Further Evaluation**
IV.B4.RP-248a		Core support shield (CSS) assembly: upper core barrel (UCB) bolt locking devices	Stainless steel; nickel alloy	Reactor coolant and neutron flux	Cracking due to fatigue	Chapter XI.M16A, “PWR Vessel Internals”	No
IV.B4.RP-248b		Core support shield (CSS) assembly: upper core barrel (UCB) bolt locking devices	Stainless steel; nickel alloy	Reactor coolant and neutron flux	Loss of material due to wear; changes in dimensions due to void swelling or distortion	Chapter XI.M16A, “PWR Vessel Internals”	No
IV.B4.RP-~~253~~252	IV.B4-~~21~~16 (R-~~191)~~188)	Core support shield (CSS) assembly: ~~(a) CSS cast outlet nozzles (Oconee Unit 3 and Davis-Besse, only); (b)~~ CSS vent valve ~~discs~~top and bottom retaining rings (valve body components)	~~Cast austenitic stainless~~Stainless steel, including CASS and PH steels	Reactor coolant and neutron flux	Loss of fracture toughness due to thermal aging embrittlement	Chapter XI.M16A, “PWR Vessel Internals” ~~Primary components (identified in the "Structure and Components" column) (for Expansion components see AMR Item IV.B4.RP-242)~~	No
IV.B4.RP-~~252~~252a	IV.B4-16 (R-188)	Core support shield (CSS) assembly: ~~(a) CSS vent valve disc shaft or hinge pin (b)~~ CSS vent valve top ~~retaining ring (c) CSS vent valve~~ and bottom retaining ~~ring~~rings; vent valve locking devices (valve body components)	Stainless steel	Reactor coolant and neutron flux	~~Loss of fracture toughness~~ Cracking due to ~~thermal aging embrittlement~~stress corrosion cracking or fatigue	Chapter XI.M16A, “PWR Vessel Internals” ~~Primary components (identified in the "Structure~~ and ~~Components" column)~~ Chapter XI.M2, “Water Chemistry” (for SCC mechanisms only) ~~No Expansion components~~	No
IV.B4.RP-251	IV.B4-15 (R-190)	Core support shield (CSS) assembly: CSS top flange	Stainless steel	Reactor coolant and neutron flux	Loss of material due to wear; loss of preload (wear)	Chapter XI.M16A, “PWR Vessel Internals”	No

IV B4	REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM Reactor Vessel Internals (PWR) – Babcock & Wilcox						
Item	**Link**	**Structure and/or Component**	**Material**	**Environment**	**Aging Effect/ Mechanism**	**Aging Management Program (AMP)**	**Further Evaluation**
IV.B4.RP-~~251~~251a	IV.B4-15 (R-190)	~~Core support shield (CSS) assembly: CSS top flange; plenum~~Plenum cover assembly: plenum cover weldment rib pads and plenum cover support flange	Stainless steel	Reactor coolant and neutron flux	Loss of material due to wear; loss of preload (wear)	Chapter XI.M16A, "PWR Vessel Internals" ~~Primary component (identified in the "Structure and Components" column) No Expansion components~~	No
IV.B4.RP-256	IV.B4-25 (R-210)	Flow distributor assembly: flow distributor bolts ~~and locking devices~~	Stainless steel; nickel alloy	Reactor coolant and neutron flux	Cracking due to stress~~-~~ corrosion cracking	~~Chapter XI.M2, "Water Chemistry," and~~ Chapter XI.M16A, "PWR Vessel Internals~~,"~~ ~~Expansion components (identified in the "Structure and Components" column)~~" and Chapter XI.M2, "Water Chemistry" ~~(for Primary components see AMR Items IV.B4.RP-247 and IV.B4.RP-248)~~	No
IV.B4.RP-256a		Flow distributor assembly: flow distributor bolt locking devices	Stainless steel; nickel alloy	Reactor coolant and neutron flux	Cracking due to fatigue	Chapter XI.M16A, "PWR Vessel Internals"	No
IV.B4.RP-256b		Flow distributor assembly: flow distributor 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	Chapter XI.M16A, "PWR Vessel Internals"	No
IV.B4.RP-259	IV.B4-31 (R-205)	Incore Monitoring ~~Instrumentation~~Instrument (IMI) guide tube assembly: ~~accessible top surfaces of~~ IMI guide tube spider-to-lower grid rib ~~sections~~section welds	Stainless steel~~; nickel alloy~~	Reactor coolant and neutron flux	Loss of fracture toughness due to ~~thermal aging,~~ neutron irradiation embrittlement	Chapter XI.M16A, "PWR Vessel Internals" ~~Primary components (identified in the "Structure and Components" column) (for Expansion components see Item IV.B4.RP-260)~~	No
IV.B4.RP-259a		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 stress corrosion cracking, irradiation-assisted stress corrosion cracking or fatigue	Chapter XI.M16A, "PWR Vessel Internals" and Chapter XI.M2, "Water Chemistry"	No.

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

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
IV.B4.RP-258	IV.B4-4 (R-183)	Incore Monitoring ~~Instrumentation~~Instrument (IMI) guide tube assembly: ~~accessible top surfaces of~~ IMI ~~Incore~~ guide tube ~~spider~~ spiders (castings~~-~~)	Cast austenitic stainless steel	Reactor coolant and neutron flux	Loss of fracture toughness due to thermal aging~~,~~ and neutron irradiation embrittlement	Chapter XI.M16A, "PWR Vessel Internals" ~~Primary components (identified in the "Structure and Components" column) (for Expansion components see Item IV.B4.RP-242)~~	No
IV.B4.RP-258a		Incore Monitoring Instrumentation (IMI) guide tube assembly: IMI guide tube spiders	Stainless steel	Reactor coolant and neutron flux	Cracking due to stress corrosion cracking, irradiation-assisted stress corrosion cracking or fatigue	Chapter XI.M16A, "PWR Vessel Internals" and Chapter XI.M2, "Water Chemistry"	No
IV.B4.RP-254	IV.B4-25 (R-210)	Lower grid assembly: alloy X-750 lower grid shock pad bolts ~~and locking devices~~ (Three Mile Island Unit ~~-~~1, only)	Nickel alloy	Reactor coolant and neutron flux	Cracking due to stress~~-~~ corrosion cracking	~~Chapter XI.M2, "Water Chemistry," and~~ Chapter XI.M16A, "PWR Vessel Internals~~,"~~ ~~Expansion components (identified in the "Structure and Components" column)~~ " and Chapter XI.M2, "Water Chemistry" ~~(for Primary components see AMR Items IV.B4.RP-247 and IV.B4.RP-248)~~	No
IV.B4.RP-254a		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	Chapter XI.M16A, "PWR Vessel Internals"	No
IV.B4.RP-254b		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	Chapter XI.M16A, "PWR Vessel Internals"	No
IV.B4.RP-246	IV.B4-12 (R-196)	Lower grid assembly: upper thermal shield (UTS) bolts and lower thermal shield (LTS) bolts	Stainless steel; nickel alloy	Reactor coolant and neutron flux	Cracking due to stress~~-~~ corrosion cracking	~~'Chapter XI.M2, "Water Chemistry," and~~ Chapter XI.M16A, "PWR Vessel Internals" and Chapter XI.M2, "Water Chemistry" ~~Expansion components (identified in the "Structure and Components" column) (for Primary components see AMR Items IV.B4.RP-247 and IV.B4.RP-248)~~	No

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

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
IV.B4.RP-246a		Lower grid assembly: upper thermal shield (UTS) bolt locking devices and lower thermal shield (LTS) bolt locking devices	Stainless steel; nickel alloy	Reactor coolant and neutron flux	Cracking due to fatigue	Chapter XI.M16A, "PWR Vessel Internals"	No
IV.B4.RP-246b		Lower grid assembly: upper thermal shield (UTS) bolt locking devices and lower thermal shield (LTS) bolt locking devices	Stainless steel; nickel alloy	Reactor coolant and neutron flux	Loss of material due to wear; changes in dimensions due to void swelling or distortion	Chapter XI.M16A, "PWR Vessel Internals"	No
IV.B4.RP-260	IV.B4-31 (R-205)	Lower grid fuel assembly: (a) ~~accessible~~ pads; (b) ~~accessible~~ pad-to-rib section welds; (c) ~~accessible~~ alloy X-750 dowels, cap screws and locking devices	Stainless steel; nickel alloy	Reactor coolant and neutron flux	Loss of fracture toughness due to neutron irradiation embrittlement	Chapter XI.M16A, "PWR Vessel Internals" ~~Expansion components (identified in the "Structure and Components" column) (for Primary components see AMR Item IV.B4.RP-259)~~	No
IV.B4.RP-260a		Lower grid fuel assembly: (a) pads; (b) pad-to-rib section welds; (c) alloy X-750 dowels, cap screws and locking devices	Stainless steel; nickel alloy	Reactor coolant and neutron flux	Cracking due to stress corrosion cracking or fatigue	Chapter XI.M16A, "PWR Vessel Internals" and Chapter XI.M2, "Water Chemistry" (for SCC mechanisms only)	No
IV.B4.RP-262	IV.B4-32 (R-203)	Lower grid assembly: ~~accessible~~ alloy X-750 dowel-to-lower fuel assembly support pad locking welds	Nickel alloy	Reactor coolant and neutron flux	Cracking due to stress~~-~~ corrosion cracking	~~Chapter XI.M2, "Water Chemistry," and~~ Chapter XI.M16A, "PWR Vessel Internals" and Chapter XI.M2, "Water Chemistry" ~~Expansion components (identified in the "Structure and Components" column) (for Primary components see AMR Item IV.B4.RP-261)~~	No
IV.B4.RP-261	IV.B4-32 (R-203)	Lower grid assembly: alloy X-750 dowel-to-guide block welds	Nickel alloy	Reactor coolant and neutron flux	Cracking due to stress~~-~~ corrosion cracking	~~Chapter XI.M2, "Water Chemistry," and~~ Chapter XI.M16A, "PWR Vessel Internals" ~~Primary components (identified in the "Structure and Components" column) (for Expansion components see AMR Items IV.B4.RP-262 and IV.B4.RP-352)~~and Chapter XI.M2, "Water Chemistry"	No

IV B4	REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM Reactor Vessel Internals (PWR) – Babcock & Wilcox						
Item	**Link**	**Structure and/or Component**	**Material**	**Environment**	**Aging Effect/ Mechanism**	**Aging Management Program (AMP)**	**Further Evaluation**
IV.B4.R-53	IV.B4-37 (R-53)	Reactor vessel internal components	Stainless steel; nickel alloy	Reactor coolant and neutron flux	Cumulative fatigue damage due to fatigue	Fatigue is a time-limited aging analysis (TLAA) to be evaluated for the period of extended operation. See the SRP, Section 4.3 “Metal Fatigue,” for acceptable methods for meeting the requirements of 10 CFR 54.21(c)(1).	Yes, TLAA
IV.B4.RP-24	IV.B4-38 (RP-24)	Reactor vessel internal components	Stainless steel; nickel alloy	Reactor coolant and neutron flux	Loss of material due to pitting and crevice corrosion	Chapter XI.M2, “Water Chemistry”	No
IV.B4.RP-376		Reactor vessel internal components	Stainless steel; nickel alloy	Reactor coolant and neutron flux	Reduction in ductility and fracture toughness due to neutron irradiation	Ductility - Reduction in Fracture Toughness is a TLAA (BAW-2248A) to be evaluated for the period of extended operation. See the SRP, Section 4.7, "Other Plant-Specific TLAAs," for acceptable methods for meeting the requirements of 10 CFR 54.21(c)(1).	Yes, TLAA
IV.B4.RP-~~238~~382	IV.B4-42 (R-179)	Reactor vessel ~~internal~~internals: ASME Section XI, Examination Category B-N-3 core support structure components ~~(inaccessible locations)~~	Stainless steel; nickel alloy	Reactor coolant and neutron flux	Cracking due to fatigue, stress corrosion cracking, ~~and~~or irradiation~~-~~assisted stress corrosion cracking; loss of material due to wear	Chapter XI.~~M2, “Water Chemistry,”~~M1, “ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD” or Chapter XI.M16A, “PWR Vessel Internals~~”~~,” by invoking applicable 10 CFR 50.55a and ASME Section XI inservice inspection requirements	~~Yes, if accessible Primary, Expansion or Existing program components indicate aging effects that need management~~No

IV B4	REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM Reactor Vessel Internals (PWR) – Babcock & Wilcox						
Item	**Link**	**Structure and/or Component**	**Material**	**Environment**	**Aging Effect/ Mechanism**	**Aging Management Program (AMP)**	**Further Evaluation**
IV.B4.RP-~~239~~352		Upper grid assembly: alloy X-750 dowel-to-upper fuel assembly support pad welds (all plants except Davis-Besse)~~Reactor vessel internal components (inaccessible locations)~~	~~Stainless steel; nickel~~Nickel alloy	Reactor coolant and neutron flux	~~Loss of fracture toughness~~Cracking due to ~~neutron irradiation embrittlement; change in dimension due to void swelling; loss of preload due to thermal and irradiation enhanced~~ stress ~~relaxation; loss of material due to wear~~ corrosion cracking	Chapter XI.M16A, "PWR Vessel Internals" and Chapter XI.M2, "Water Chemistry"	~~Yes, if accessible Primary, Expansion or Existing program components indicate aging effects that need management~~No
IV.B4.RP-236		Reactor internal "No Additional Measures" components~~Reactor vessel internal components with no additional measures~~	Stainless steel; nickel alloy	Reactor coolant and neutron flux	No additional aging management for reactor internal "No Additional Measures" components unless required by ASME Section XI, Examination Category B-N-3 or relevant operating experience exists~~Cracking due to stress corrosion cracking, and irradiation-assisted stress corrosion cracking~~	~~Chapter XI.M2, "Water Chemistry" and~~ Chapter XI.M16A, "PWR Vessel Internals" ~~Note: Components with no additional measures are not uniquely identifies in GALL tables - Components with no additional measures are defined in Section 3.3.1 of MRP-227, "Materials Reliability Program: Pressurized Water Reactor Internals Inspection and Evaluation Guidelines"~~	No
IV.B4.RP-400		Core support shield assembly: upper (top) flange weld	Stainless steel	Reactor coolant and neutron flux	Cracking due to stress-corrosion cracking	Chapter XI.M16A, "PWR Vessel Internals" and Chapter XI.M2, "Water Chemistry"	No

IV B4	REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM Reactor Vessel Internals (PWR) – Babcock & Wilcox						
Item	**Link**	**Structure and/or Component**	**Material**	**Environment**	**Aging Effect/ Mechanism**	**Aging Management Program (AMP)**	**Further Evaluation**
IV.B4.RP-~~237~~401		~~Reactor vessel internal components with no additional measures~~Core support shield assembly: upper (top) flange weld	Stainless steel~~; nickel alloy~~	Reactor coolant and neutron flux	Loss of fracture toughness due to neutron irradiation embrittlement~~; change in dimension due to void swelling; loss of preload due to thermal and irradiation enhanced stress relaxation; loss of material due to wear~~	Chapter XI.M16A, "PWR Vessel Internals" ~~Note: Components with no additional measures are not uniquely identified in GALL tables - Components with no additional measures are defined in Section 3.3.1 of MRP-227, "Materials Reliability Program: Pressurized Water Reactor Internals Inspection and Evaluation Guidelines"~~	No
~~IV.B4.RP-382~~	~~IV.B4-42(R-179)~~	~~Reactor vessel internals: core support structure~~	~~Stainless steel; nickel alloy; cast austenitic stainless steel~~	~~Reactor coolant and neutron flux~~	~~Cracking, or Loss of material due to wear~~	~~Chapter XI.M1, "ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD"~~	~~No~~
~~IV.B4.RP-352~~		~~Upper grid assembly: alloy X-750 dowel to upper fuel assembly support pad welds (all plants except Davis-Besse)~~	~~Nickel alloy~~	~~Reactor coolant and neutron flux~~	~~Cracking due to stress corrosion cracking~~	~~Chapter XI.M2, "Water Chemistry," and Chapter XI.M16A, "PWR Vessel Internals" Expansion components (identified in the "Structure and Components" column) (for Primary components see AMR Item IV.B4.RP-261)~~	~~No~~

(5) Mark-up of changes to GALL Report Chapter IX.C and IX.G

IX.C Selected Definitions & Use of Terms for Describing and Standardizing MATERIALS

Stainless steel	Products grouped under the term "stainless steel" include ~~wrought or forged~~ austenitic, ferritic, martensitic, precipitation-hardened (PH), or duplex stainless steel (Cr content >11%). These stainless steels may be fabricated using a wrought or cast process. These materials are susceptible to a variety of aging effects and mechanisms, including loss of material due to pitting and crevice corrosion, and cracking due to stress ~~-~~corrosion cracking. In some cases, when ~~the recommended AMP~~an aging effect is applicable to all of the ~~same for PH~~various stainless steel ~~or cast~~categories, it can be assumed that the term "stainless steel" in the "Material" column of an AMR line-item in the GALL Report encompasses all stainless steel types. Cast austenitic stainless steel (CASS) ~~as for stainless steel, PH stainless steel or CASS are included as a part of the stainless steel classification. However, CASS~~ is quite susceptible to loss of fracture toughness due to thermal and neutron irradiation embrittlement. ~~Therefore, when this aging effect is being considered, CASS~~ In addition, MRP-227-A indicates that PH stainless steels or martensitic stainless steels may be susceptible to loss of fracture toughness by a thermal aging mechanism. Therefore, when loss of fracture toughness due to thermal and neutron irradiation embrittlement is an applicable aging effect and mechanism for a component in the GALL Report, the CASS, PH stainless steel, or martensitic stainless steel designation is specifically identified ~~designated~~ in an AMR line-item. Steel with stainless steel cladding also may be considered stainless steel when the aging effect is associated with the stainless steel surface of the material, rather than the composite volume of the material. Examples of stainless steel designations that comprise this category include A-286, SA193-Gr. B8, SA193-Gr. B8M, Gr. 660 (A-286), SA193-6, SA193-Gr. B8 or B-8M, SA453, Type 416, Type 403, 410, 420, and ~~Types~~431 martensitic stainless steels, Type 15-5, 17-4, and 13-8-Mo PH stainless steels, and SA-193, Grade B8 and B8M bolting materials. Examples of wrought austenitic stainless materials that comprise this category include Type 304, 304NG, 304L, 308, 308L, 309, 309L, 316~~,~~ and 347~~, 403, and 416.~~. Examples of CASS ~~designations~~that comprise this category

	include ~~CF-3, -8, -3M,~~CF3, CF3M, CF8 and ~~-8M.~~CF8M. [Ref. 6, 7~~]~~, 30]

IX.G References

30. Welding Handbook, Seventh Edition, Volume 4, Metals and Their Weldability, American Welding Society, 1984, p. 76-145.

Appendix B, Section 2 – Mark-up of Changes to the SRP-LR

In the mark-up, red or green strikethrough text indicates a deletion and blue underline text indicates an insertion. Green text indicates a move, where a double strikethrough indicates the original location of the text and a double underline indicates the final location of the moved text.

(1) Mark-up of changes to SRP-LR Table 3.0-1

Table 3.0-1 FSAR Supplement for Aging Management of Applicable Systems				
GALL Chapter	**GALL Program**	**Description of Program**	**Implementation Schedule**	**Applicable GALL Report and SRP-LR Chapter References**
XI.M16A	PWR Vessel Internals	The program relies on implementation of the inspection and evaluation guidelines in EPRI Technical Report No. ~~10165961~~022863 (MRP-227-A) and EPRI Technical Report No. 1016609 (MRP-228) to manage the aging effects on the reactor vessel internal components. This program is used to manage (a) ~~various forms of~~ cracking, including stress corrosion cracking~~SCC~~, primary water stress corrosion cracking~~PWSCC~~, irradiation-assisted stress -corrosion cracking ~~(IASCC)~~, ~~or~~and cracking due to fatigue/cyclical loading; (b) loss of material induced by wear; (c) loss of fracture toughness due to either thermal aging ~~or~~, neutron irradiation embrittlement, or void swelling; (d) dimensional changes ~~and potential loss of fracture toughness~~ due to void swelling ~~and irradiation growth~~or distortion; and (e) loss of preload due to thermal and irradiation-enhanced stress relaxation or creep.	Program should be implemented prior to period of extended operation	GALL IV / SRP 3.1

(2) Mark-up of changes to SRP-LR Section 3.1.2, "Acceptance Criteria"

3.1.2.2.9 *Removed as a result of LR-ISG-2011-04*~~**Cracking due to Stress Corrosion Cracking and Irradiation-Assisted Stress Corrosion Cracking**~~

~~Cracking due to SCC and irradiation-assisted stress corrosion cracking (IASCC) could occur in inaccessible locations for stainless steel and nickel-alloy Primary and Expansion PWR reactor vessel internal components. If aging effects are identified in accessible locations, the GALL Report recommends further evaluation of the aging effects in inaccessible locations on a plant-specific basis to ensure that this aging effect is adequately managed. Acceptance criteria are described in Branch Technical Position RLSB-1 (Appendix A.1 of this SRP-LR).~~

3.1.2.2.10 *Removed as a result of LR-ISG-2011-04*~~**Loss of Fracture Toughness due to Neutron Irradiation Embrittlement; Change in Dimension due to Void Swelling; Loss of Preload due to Stress Relaxation; or Loss of Material due to Wear**~~

~~Loss of fracture toughness due to neutron irradiation embrittlement, change in dimension due to void swelling, loss of preload due to stress relaxation, or loss of material due to wear could occur in inaccessible locations for stainless steel and nickel-alloy Primary and Expansion PWR reactor vessel internal components. If aging effects are identified in accessible locations, the GALL Report recommends further evaluation of the aging effects in inaccessible locations on a plant-specific basis to ensure that this aging effect is adequately managed. Acceptance criteria are described in Branch Technical Position RLSB-1 (Appendix A.1 of this SRP-LR).~~

3.1.2.2.12 *Removed as a result of LR-ISG-2011-04*~~**Cracking due to Fatigue**~~

~~EPRI 1016596, *Materials Reliability Program: Pressurized Water Reactor Internals Inspection and Evaluation Guidelines* (MRP-227-Rev. 0) identifies cracking due to fatigue as an aging effect that can occur for the lower flange weld in the core support barrel assembly, fuel alignment plate in the upper internals assembly, and core support plate lower support structure in PWR internals designed by Combustion Engineering. The GALL Report recommends that inspection for cracking in this component be performed if acceptable fatigue life cannot be demonstrated by TLAA through the period of extended operation as defined in 10 CFR 54.3.~~

3.1.2.2.13 *Removed as a result of LR-ISG-2011-04*~~**Cracking due to Stress Corrosion Cracking and Fatigue**~~

~~Cracking due to stress corrosion cracking and fatigue could occur in nickel alloy control rod guide tube assemblies, guide tube support pins exposed to reactor coolant, and neutron flux. The GALL Report, AMR Item IV.B2.RP-355, recommends further evaluation of a plant-specific AMP to ensure this aging effect is adequately managed. Acceptance criteria are described in Branch Technical Position RLSB-1 (Appendix A.1 of this SRP-LR).~~

3.1.2.2.14 *Removed as a result of LR-ISG-2011-04*~~**Loss of Material due to Wear**~~

~~Loss of material due to wear could occur in nickel alloy control rod guide tube assemblies, guide tube support pins and in Zircaloy-4 incore instrumentation lower thimble tubes exposed to reactor coolant, and neutron flux. The GALL Report, AMR Items IV.B2.RP-356 and IV.B3.RP-357, recommends further evaluation of a plant-specific AMP to ensure this aging effect is adequately managed. Acceptance criteria are described in Branch Technical Position RLSB-1 (Appendix A.1 of this SRP-LR).~~

(3) Mark-up of changes to SRP-LR Section 3.1.3, "Review Procedures"

3.1.3.2.9 *Removed as a result of LR-ISG-2011-04*~~**Cracking due to Stress Corrosion Cracking and Irradiation-Assisted Stress Corrosion Cracking**~~

~~The GALL Report recommends further evaluation of cracking due to SCC and IASCC for inaccessible locations for Primary and Expansion PWR reactor vessel internal components if aging effects are identified for these components in accessible locations. The reviewer reviews the applicant's proposed program on a case-by-case basis to ensure that an adequate program~~

~~will be in place for the management of these aging effects consistent with the action item documented in the staff's safety evaluation for MRP-227, Revision 0..~~

3.1.3.2.10 *Removed as a result of LR-ISG-2011-04* ~~**Loss of Fracture Toughness due to Neutron Irradiation Embrittlement; Change in Dimension due to Void Swelling; Loss of Preload due to Stress Relaxation; or Loss of Material due to Wear**~~

~~The GALL Report recommends further evaluation of loss of fracture toughness due to neutron irradiation embrittlement, change in dimension due to void swelling, loss of preload due to stress relaxation, or loss of material due to wear for inaccessible locations for Primary and Expansion PWR reactor vessel internal components, if aging effects are identified for these components in accessible locations. The reviewer reviews the applicant's proposed program on a case-by-case basis to ensure that an adequate program will be in place for the management of these aging effects consistent with the action item documented in the staff's safety evaluation for MRP-227, Revision 0.~~

3.1.3.2.12 *Removed as a result of LR-ISG-2011-04* ~~**Cracking due to Fatigue**~~

~~The GALL Report recommends further evaluation of cracking due to fatigue in the lower flange weld in the core support barrel assembly, fuel alignment plate in the upper internals assembly, and core support plate in the lower support structure in PWR internals designed by Combustion Engineering. The reviewer determines whether a TLAA has been performed for each component, consistent with the action item documented in the staff's safety evaluation for MRP-227, Revision 0. If a TLAA has not been performed, the reviewer determines whether the applicant has performed an evaluation to identify the potential location and extent of fatigue cracking for each component consistent with the action item documented in the staff's safety evaluation for MRP-227, Revision 0.~~

3.1.3.2.13 *Removed as a result of LR-ISG-2011-04* ~~**Cracking due to Stress Corrosion Cracking and Fatigue**~~

~~The GALL Report recommends further evaluation of cracking due to stress corrosion cracking and fatigue in the nickel alloy control rod guide tube assemblies, guide tube support pins exposed to reactor coolant, and neutron flux. The reviewer reviews the applicant's proposed program on a case-by-case basis to ensure that an adequate program will be in place for the management of these aging effects consistent with the action item documented in the staff's safety evaluation for MRP-227, Revision 0.~~

3.1.3.2.14 *Removed as a result of LR-ISG-2011-04* ~~**Loss of Material due to Wear**~~

~~The GALL Report recommends further evaluation of loss of material due to wear in nickel alloy control rod guide tube assemblies, guide tube support pins and in Zircaloy-4 incore instrumentation lower thimble tubes exposed to reactor coolant, and neutron flux. The reviewer reviews the applicant's proposed program on a case-by-case basis to ensure that an adequate program will be in place for the management of these aging effects consistent with the action item documented in the staff's safety evaluation for MRP-227, Revision 0.~~

(4) Mark-up of changes to SRP-LR 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 Report							
ID	**Type**	**Component**	**Aging Effect/Mechanism**	**Aging Management Programs**	**Further Evaluation Recommended**	**Rev2 Item**	**Rev1 Item**
3	BWR/ PWR	Stainless steel or nickel alloy reactor vessel internal components exposed to reactor coolant and neutron flux	Cumulative fatigue damage due to fatigue	Fatigue is a TLAA evaluated for the period of extended operation (See SRP, Section 4.3 "Metal Fatigue," for acceptable methods to comply with 10 CFR 54.21(c)(1)	Yes, TLAA (See subsection 3.1.2.2.1)	IV.B1.R-53 IV.B2.RP-303 IV.B3.RP-339 IV.B4.R-53 ~~IV.B3.RP-389~~ ~~IV.B3.RP-390~~ ~~IV.B3.RP-391~~	IV.B1-14 (R-53) IV.B2-31 (R-53) IV.B3-24 (R-53) IV.B4-37 (R-53) ~~N/A~~ ~~N/A~~ ~~N/A~~
15	PWR	Stainless steel Babcock & Wilcox (including CASS, martensitic SS, and PH SS) and nickel alloy reactor vessel internal components exposed to reactor coolant and neutron flux	Reduction ~~in~~of ductility and fracture toughness due to neutron irradiation embrittlement, and for CASS, martensitic SS, and PH SS due to thermal aging embrittlement	Ductility - Reduction in ~~f~~Fracture ~~t~~Toughness is a TLAA to be evaluated for the period of extended operation~~.~~, See the SRP, Section 4.7, "Other Plant-Specific TLAAs," for acceptable methods ~~for~~of meeting the requirements of 10 CFR 54.21(c~~)(1).~~).	Yes, TLAA (See subsection 3.1.2.2.3.3)	IV.B4.RP-376	N/A
~~23~~	~~PWR~~	~~Stainless steel or nickel alloy PWR reactor vessel internal components (inaccessible locations) exposed to reactor coolant and neutron flux~~	~~Cracking due to stress corrosion cracking, and irradiation-assisted stress corrosion cracking~~	~~Chapter XI.M16A, "PWR Vessel Internals," and Chapter XI.M2, "Water Chemistry"~~	~~Yes, if accessible Primary, Expansion or Existing program components indicate aging effects that need management (See subsection 3.1.2.2.9)~~	~~IV.B2.RP-268~~ ~~IV.B3.RP-309~~ ~~IV.B4.RP-238~~	~~N/A~~ ~~N/A~~ ~~N/A~~
~~24~~	~~PWR~~	~~Stainless steel or nickel alloy PWR reactor vessel internal components (inaccessible locations) exposed to reactor coolant and neutron flux~~	~~Loss of fracture toughness due to neutron irradiation embrittlement; or changes in dimension due to void swelling; or loss of preload due to thermal and irradiation enhanced stress relaxation; or loss of material due to wear~~	~~Chapter XI.M16A, "PWR Vessel Internals"~~	~~Yes, if accessible Primary, Expansion or Existing program components indicate aging effects that need management (See subsection 3.1.2.2.10)~~	~~IV.B2.RP-269~~ ~~IV.B3.RP-311~~ ~~IV.B4.RP-239~~	~~N/A~~ ~~N/A~~ ~~N/A~~

Table 3.1-1 Summary of Aging Management Programs for Reactor Vessel, Internals, and Reactor Coolant System Evaluated in Chapter IV of the GALL Report							
ID	**Type**	**Component**	**Aging Effect/Mechanism**	**Aging Management Programs**	**Further Evaluation Recommended**	**Rev2 Item**	**Rev1 Item**
~~26~~	~~PWR~~	~~Stainless steel Combustion Engineering core support barrel assembly: lower flange weld exposed to reactor coolant and neutron flux; Upper internals assembly: fuel alignment plate (applicable to plants with core shrouds assembled with full height shroud plates) exposed to reactor coolant and neutron flux; Lower support structure: core support plate (applicable to plants with a core support plate) exposed to reactor coolant and neutron flux~~	~~Cracking due to fatigue~~	~~Chapter XI.M16A, "PWR Vessel Internals," and Chapter XI.M2, "Water Chemistry," if fatigue life cannot be confirmed by TLAA~~	~~Yes, evaluate to determine the potential locations and extent of fatigue cracking (See subsection 3.1.2.2.12)~~	~~IV.B3.RP-333~~ ~~IV.B3.RP-338~~ ~~IV.B3.RP-343~~	~~N/A~~
~~27~~	~~PWR~~	~~Nickel alloy Westinghouse control rod guide tube assemblies, guide tube support pins exposed to reactor coolant and neutron flux~~	~~Cracking due to stress corrosion cracking and fatigue~~	~~A plant-specific aging management program is to be evaluated~~	~~Yes, plant-specific (See subsection 3.1.2.2.13)~~	~~IV.B2.RP-355~~	~~N/A~~
~~28~~	~~PWR~~	~~Nickel alloy Westinghouse control rod guide tube assemblies, guide tube support pins, and Zircaloy-4 Combustion Engineering incore instrumentation thimble tubes exposed to reactor coolant and neutron flux~~	~~Loss of material due to wear~~	~~A plant-specific aging management program is to be evaluated~~	~~Yes, plant-specific (See subsection 3.1.2.2.14)~~	~~IV.B2.RP-356~~ ~~IV.B3.RP-357~~	~~N/A~~ ~~N/A~~
28	PWR	Stainless steel Combustion Engineering "Existing Programs" components exposed to reactor coolant and neutron flux	Loss of material due to wear; cracking due to stress corrosion cracking, irradiation-assisted stress corrosion cracking, or fatigue	Chapter XI.M16A, "PWR Vessel Internals," and Chapter XI.M2, "Water Chemistry" (for SCC mechanisms only)	No	IV.B3.RP-400	N/A

Table 3.1-1 Summary of Aging Management Programs for Reactor Vessel, Internals, and Reactor Coolant System Evaluated in Chapter IV of the GALL Report							
ID	**Type**	**Component**	**Aging Effect/Mechanism**	**Aging Management Programs**	**Further Evaluation Recommended**	**Rev2 Item**	**Rev1 Item**
32	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, or loss of material due to wear	Chapter XI.M1, "ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD" or Chapter XI.M16A, "PWR Vessel Internals," invoking applicable 10 CFR 50.55a and ASME Section XI inservice inspection requirements for these components	No	IV.B2.RP-382 IV.B3.RP-382 IV.B4.RP-382	IV.B2-26 (R-142) IV.B3-22 (R-170) IV.B4-42 (R-179)
~~51~~	~~PWR~~	~~Stainless steel or nickel-alloy Babcock & Wilcox reactor internal components exposed to reactor coolant and neutron flux~~	~~Cracking due to stress corrosion cracking, irradiation-assisted stress corrosion cracking, or fatigue~~	~~Chapter XI.M16A, "PWR Vessel Internals," and Chapter XI.M2, "Water Chemistry"~~	~~No~~	~~IV.B4.RP-236~~ ~~IV.B4.RP-241~~ ~~IV.B4.RP-244~~ ~~IV.B4.RP-245~~ ~~IV.B4.RP-246~~ ~~IV.B4.RP-247~~ ~~IV.B4.RP-248~~ ~~IV.B4.RP-254~~ ~~IV.B4.RP-256~~ ~~IV.B4.RP-261~~ ~~IV.B4.RP-262~~ ~~IV.B4.RP-352~~ ~~IV.B4.RP-375~~	~~N/A~~ ~~IV.B4-7(R-125)~~ ~~IV.B4-7(R-125)~~ ~~IV.B4-13(R-194)~~ ~~IV.B4-12(R-196)~~ ~~IV.B4-13(R-194)~~ ~~IV.B4-12(R-196)~~ ~~IV.B4-25(R-210)~~ ~~IV.B4-25(R-210)~~ ~~IV.B4-32(R-203)~~ ~~IV.B4-32(R-203)~~ ~~N/A~~ ~~N/A~~
~~52~~	~~PWR~~	~~Stainless steel or nickel-alloy Combustion Engineering reactor internal components exposed to reactor coolant and neutron flux~~	~~Cracking due to stress corrosion cracking, irradiation-assisted stress corrosion cracking, or fatigue~~	~~Chapter XI.M16A, "PWR Vessel Internals," and Chapter XI.M2, "Water Chemistry"~~	~~No~~	~~IV.B3.RP-306~~ ~~IV.B3.RP-312~~ ~~IV.B3.RP-313~~ ~~IV.B3.RP-314~~ ~~IV.B3.RP-316~~ ~~IV.B3.RP-320~~ ~~IV.B3.RP-322~~ ~~IV.B3.RP-323~~ ~~IV.B3.RP-324~~ ~~IV.B3.RP-325~~ ~~IV.B3.RP-327~~ ~~IV.B3.RP-328~~ ~~IV.B3.RP-329~~ ~~IV.B3.RP-330~~ ~~IV.B3.RP-334~~ ~~IV.B3.RP-335~~ ~~IV.B3.RP-342~~ ~~IV.B3.RP-358~~	~~N/A~~ ~~IV.B3-2(R-149)~~ ~~N/A~~ ~~IV.B3-9(R-162)~~ ~~IV.B3-9(R-162)~~ ~~IV.B3-9(R-162)~~ ~~N/A~~ ~~N/A~~ ~~N/A~~ ~~N/A~~ ~~IV.B3-15(R-155)~~ ~~IV.B3-15(R-155)~~ ~~IV.B3-15(R-155)~~ ~~IV.B3-23(R-167)~~ ~~IV.B3-23(R-167)~~ ~~IV.B3-23(R-167)~~ ~~N/A~~ ~~N/A~~

Table 3.1-1 Summary of Aging Management Programs for Reactor Vessel, Internals, and Reactor Coolant System Evaluated in Chapter IV of the GALL Report							
ID	**Type**	**Component**	**Aging Effect/Mechanism**	**Aging Management Programs**	**Further Evaluation Recommended**	**Rev2 Item**	**Rev1 Item**
~~53~~	~~PWR~~	~~Stainless steel or nickel alloy Westinghouse reactor internal components exposed to reactor coolant and neutron flux~~	~~Cracking due to stress corrosion cracking, irradiation-assisted stress corrosion cracking, or fatigue~~	~~Chapter XI.M16A, "PWR Vessel Internals," and Chapter XI.M2, "Water Chemistry"~~	~~No~~	~~IV.B2.RP-265~~ ~~IV.B2.RP-271~~ ~~IV.B2.RP-273~~ ~~IV.B2.RP-275~~ ~~IV.B2.RP-276~~ ~~IV.B2.RP-278~~ ~~IV.B2.RP-280~~ ~~IV.B2.RP-282~~ ~~IV.B2.RP-286~~ ~~IV.B2.RP-289~~ ~~IV.B2.RP-291~~ ~~IV.B2.RP-293~~ ~~IV.B2.RP-294~~ ~~IV.B2.RP-298~~ ~~IV.B2.RP-301~~ ~~IV.B2.RP-346~~ ~~IV.B2.RP-387~~	~~N/A~~ ~~IV.B2-10(R-125)~~ ~~IV.B2-10(R-125)~~ ~~IV.B2-6(R-128)~~ ~~IV.B2-8(R-120)~~ ~~IV.B2-8(R-120)~~ ~~IV.B2-8(R-120)~~ ~~IV.B2-8(R-120)~~ ~~IV.B2-16(R-133)~~ ~~IV.B2-20(R-130)~~ ~~IV.B2-24(R-138)~~ ~~IV.B2-24(R-138)~~ ~~IV.B2-24(R-138)~~ ~~IV.B2-28(R-118)~~ ~~IV.B2-40(R-112)~~ ~~N/A~~ ~~N/A~~
51a	PWR	Stainless steel or nickel alloy Babcock & Wilcox reactor internal "Primary" components exposed to reactor coolant and neutron flux	Cracking due to stress corrosion cracking, irradiation-assisted stress corrosion cracking, or fatigue	Chapter XI.M16A, "PWR Vessel Internals," and Chapter XI.M2, "Water Chemistry" (for SCC mechanisms only)	No	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-256 IV.B4.RP-256a IV.B4.RP-258a IV.B4.RP-259a IV.B4.RP-261 IV.B4.RP-400	IV.B4-7 (R-125) N/A N/A IV.B4-13 (R-194) N/A IV.B4-25 (R-210) N/A N/A N/A IV.B4-25 (R-210) N/A N/A N/A IV.B4-32 (R-203) N/A
51b	PWR	Stainless steel or nickel alloy Babcock & Wilcox reactor internal "Expansion" components exposed to reactor coolant and neutron flux	Cracking due to stress corrosion cracking, irradiation-assisted stress corrosion cracking, fatigue, or overload	Chapter XI.M16A, "PWR Vessel Internals," and Chapter XI.M2, "Water Chemistry" (for SCC mechanisms only)	No	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-254 IV.B4.RP-254a	IV.B4-7 (R-125) N/A IV.B4-13 (R-194) N/A IV.B4-12 (R-196) N/A IV.B4-25 (R-210) N/A

Table 3.1-1 Summary of Aging Management Programs for Reactor Vessel, Internals, and Reactor Coolant System Evaluated in Chapter IV of the GALL Report							
ID	**Type**	**Component**	**Aging Effect/Mechanism**	**Aging Management Programs**	**Further Evaluation Recommended**	**Rev2 Item**	**Rev1 Item**
						IV.B4.RP-260a IV.B4.RP-262 IV.B4.RP-352 IV.B4.RP-250a IV.B4.RP-375	N/A IV.B4-32 (R-203) N/A N/A N/A
52a	PWR	Stainless steel or nickel alloy Combustion Engineering reactor internal "Primary" components exposed to reactor coolant and neutron flux	Cracking due to stress corrosion cracking, irradiation-assisted stress corrosion cracking, or fatigue	Chapter XI.M16A, "PWR Vessel Internals," and Chapter XI.M2, "Water Chemistry" (for SCC mechanisms only)	No	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-342 IV.B3.RP-358 IV.B3.RP-362a IV.B3.RP-363 IV.B3.RP-338 IV.B3.RP-343	IV.B3-2 (R-149) IV.B3-9 (R-162) N/A N/A N/A IV.B3-15 (R-155) IV.B3-15 (R-155) N/A N/A N/A N/A N/A N/A
52b	PWR	Stainless steel or nickel alloy Combustion Engineering reactor internal "Expansion" components exposed to reactor coolant and neutron flux	Cracking due to stress corrosion cracking, irradiation-assisted stress corrosion cracking, or fatigue	Chapter XI.M16A, "PWR Vessel Internals," and Chapter XI.M2, "Water Chemistry" (for SCC mechanisms only)	No	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	NA IV.B3-9 (R-162) N/A N/A IV.B3-12 (R-155) IV.B3-23 (R-167) N/A IV.B3-23 (R-167) N/A
52c	PWR	Stainless steel or nickel alloy Combustion Engineering reactor internal "Existing Programs" components exposed to reactor coolant and neutron flux	Cracking due to stress corrosion cracking, irradiation-assisted stress corrosion cracking, or fatigue	Chapter XI.M16A, "PWR Vessel Internals," and Chapter XI.M2, "Water Chemistry" (for SCC mechanisms only)	No	IV.B3.RP-320 IV.B3.RP-334	IV.B3-9 (R-162) IV.B3-23 (R-167)

Table 3.1-1 Summary of Aging Management Programs for Reactor Vessel, Internals, and Reactor Coolant System Evaluated in Chapter IV of the GALL Report							
ID	**Type**	**Component**	**Aging Effect/Mechanism**	**Aging Management Programs**	**Further Evaluation Recommended**	**Rev2 Item**	**Rev1 Item**
53a	PWR	Stainless steel or nickel alloy Westinghouse reactor internal "Primary" components exposed to reactor coolant and neutron flux	Cracking due to stress corrosion cracking, irradiation-assisted stress corrosion cracking, or fatigue	Chapter XI.M16A, "PWR Vessel Internals," and Chapter XI.M2, "Water Chemistry" (for SCC mechanisms only)	No	IV.B2.RP-270a IV.B2.RP-271 IV.B2.RP-275 IV.B2.RP-276 IV.B2.RP-280 IV.B2.RP-298 IV.B2.RP-302 IV.B2.RP-387	N/A IV.B2-10 (R-125) IV.B2-6 (R-128) IV.B2-8 (R-120) IV.B2-8 (R-120) IV.B2-28 (R-118) N/A N/A
53b	PWR	Stainless steel Westinghouse reactor internal "Expansion" components exposed to reactor coolant and neutron flux	Cracking due to stress corrosion cracking, irradiation-assisted stress corrosion cracking, or fatigue	Chapter XI.M16A, "PWR Vessel Internals," and Chapter XI.M2, "Water Chemistry" (for SCC mechanisms only)	No	IV.B2.RP-273 IV.B2.RP-278 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-387a	IV.B2-10 (R-125) IV.B2-8 (R-120) IV.B2-16 (R-133) IV.B2-24 (R-138) N/A N/A IV.B2-24 (R-138) IV.B2-24 (R-138) N/A
53c	PWR	Stainless steel or nickel alloy Westinghouse reactor internal "Existing Programs" components exposed to reactor coolant and neutron flux	Cracking due to stress corrosion cracking, irradiation-assisted stress corrosion cracking, or fatigue	Chapter XI.M16A, "PWR Vessel Internals," and Chapter XI.M2, "Water Chemistry" (for SCC mechanisms only)	No	IV.B2.RP-289 IV.B2.RP-301 IV.B2.RP-345 IV.B2.RP-346 IV.B2.RP-399 IV.B2.RP-355	IV.B2-20 (R-130) IV.B2-40 (R-112) N/A N/A N/A N/A
54	PWR	Stainless steel bottom mounted instrument system flux thimble tubes (with or without chrome plating) exposed to reactor coolant and neutron flux (Westinghouse "Existing Programs" components)	Loss of material due to wear	Chapter XI.M16A, "PWR Vessel Internals," ~~and~~ or Chapter XI.M37, ~~"~~"Flux Thimble Tube Inspection~~"~~"	No	IV.B2.RP-284	~~IV.B2-12(R-143)~~ IV.B2-13 (R-145)
~~55~~	~~PWR~~	~~Stainless steel thermal shield assembly, thermal shield flexures exposed to reactor coolant and neutron flux~~	~~Cracking due to fatigue; Loss of material due to wear~~	~~Chapter XI.M16A, "PWR Vessel Internals"~~	~~No~~	~~IV.B2.RP-302~~	~~N/A~~
55a	PWR	Stainless steel or nickel alloy Babcock and Wilcox reactor internal "No Additional Measures"	No additional aging management for reactor internal "No Additional Measures" components	Chapter XI.M16A, "PWR Vessel Internals"	No	IV.B4.RP-236	NA

Table 3.1-1 Summary of Aging Management Programs for Reactor Vessel, Internals, and Reactor Coolant System Evaluated in Chapter IV of the GALL Report							
ID	**Type**	**Component**	**Aging Effect/Mechanism**	**Aging Management Programs**	**Further Evaluation Recommended**	**Rev2 Item**	**Rev1 Item**
		components exposed to reactor coolant and neutron flux	unless required by ASME Section XI, Examination Category B-N-3 or relevant operating experience invalidates MRP-227-A.				
55b	PWR	Stainless steel or nickel alloy Combustion Engineering reactor internal "No Additional Measures" components exposed to reactor coolant and neutron flux	No additional aging management for reactor internal "No Additional Measures" components unless required by ASME Section XI, Examination Category B-N-3 or relevant operating experience invalidates MRP-227-A.	Chapter XI.M16A, "PWR Vessel Internals"	No	IV.B3.RP-306	NA
55c	PWR	Stainless steel or nickel alloy Westinghouse reactor internal "No Additional Measures" components exposed to reactor coolant and neutron flux	No additional aging management for reactor internal "No Additional Measures" components unless required by ASME Section XI, Examination Category B-N-3 or relevant operating experience invalidates MRP-227-A.	Chapter XI.M16A, "PWR Vessel Internals"	No	IV.B2.RP-265	NA
~~56~~	~~PWR~~	~~Stainless steel or nickel-alloy Combustion Engineering reactor internal components exposed to reactor coolant and neutron flux~~	~~Loss of fracture toughness due to neutron irradiation embrittlement; or changes in dimension due to void swelling; or loss of preload due to thermal and irradiation enhanced stress relaxation; or loss of material due to wear~~	~~Chapter XI.M16A, "PWR Vessel Internals"~~	~~No~~	~~IV.B3.RP-307~~ ~~IV.B3.RP-315~~ ~~IV.B3.RP-317~~ ~~IV.B3.RP-318~~ ~~IV.B3.RP-319~~ ~~IV.B3.RP-326~~ ~~IV.B3.RP-331~~ ~~IV.B3.RP-332~~ ~~IV.B3.RP-336~~ ~~IV.B3.RP-359~~ ~~IV.B3.RP-360~~ ~~IV.B3.RP-361~~ ~~IV.B3.RP-362~~ ~~IV.B3.RP-363~~ ~~IV.B3.RP-364~~ ~~IV.B3.RP-365~~	~~N/A~~ ~~IV.B3-7(R-165)~~ ~~IV.B3-7(R-165)~~ ~~IV.B4-8(R-163)~~ ~~IV.B3-9(R-162)~~ ~~N/A~~ ~~N/A~~ ~~IV.B3-17(R-156)~~ ~~IV.B3-22(R-170)~~ ~~N/A~~ ~~N/A~~ ~~N/A~~ ~~N/A~~ ~~N/A~~ ~~N/A~~ ~~N/A~~

Table 3.1-1 Summary of Aging Management Programs for Reactor Vessel, Internals, and Reactor Coolant System Evaluated in Chapter IV of the GALL Report							
ID	**Type**	**Component**	**Aging Effect/Mechanism**	**Aging Management Programs**	**Further Evaluation Recommended**	**Rev2 Item**	**Rev1 Item**
						~~IV.B3.RP-366~~	~~N/A~~
~~58~~	~~PWR~~	~~Stainless steel or nickel-alloy Babcock & Wilcox reactor internal components exposed to reactor coolant and neutron flux~~	~~Loss of fracture toughness due to neutron irradiation embrittlement; or changes in dimension due to void swelling; or loss of preload due to thermal and irradiation enhanced stress relaxation; or loss of material due to wear~~	~~Chapter XI.M16A, "PWR Vessel Internals"~~	~~No~~	~~IV.B4.RP-237~~ ~~IV.B4.RP-240~~ ~~IV.B4.RP-242~~ ~~IV.B4.RP-243~~ ~~IV.B4.RP-249~~ ~~IV.B4.RP-250~~ ~~IV.B4.RP-251~~ ~~IV.B4.RP-252~~ ~~IV.B4.RP-253~~ ~~IV.B4.RP-258~~ ~~IV.B4.RP-259~~ ~~IV.B4.RP-260~~	~~N/A~~ ~~IV.B4-1(R-128)~~ ~~IV.B4-4(R-183)~~ ~~IV.B4-1(R-128)~~ ~~IV.B4-12(R-196)~~ ~~IV.B4-12(R-196)~~ ~~IV.B4-15(R-190)~~ ~~IV.B4-16(R-188)~~ ~~IV.B4-21(R-191)~~ ~~IV.B4-4(R-183)~~ ~~IV.B4-31(R-205)~~ ~~IV.B4-31(R-205)~~
~~59~~	~~PWR~~	~~Stainless steel or nickel-alloy Westinghouse reactor internal components exposed to reactor coolant and neutron flux~~	~~Loss of fracture toughness due to neutron irradiation embrittlement; or changes in dimension due to void swelling; or loss of preload due to thermal and irradiation enhanced stress relaxation; or loss of material due to wear~~	~~Chapter XI.M16A, "PWR Vessel Internals"~~	~~No~~	~~IV.B2.RP-267~~ ~~IV.B2.RP-270~~ ~~IV.B2.RP-272~~ ~~IV.B2.RP-274~~ ~~IV.B2.RP-281~~ ~~IV.B2.RP-285~~ ~~IV.B2.RP-287~~ ~~IV.B2.RP-288~~ ~~IV.B2.RP-290~~ ~~IV.B2.RP-292~~ ~~IV.B2.RP-295~~ ~~IV.B2.RP-296~~ ~~IV.B2.RP-297~~ ~~IV.B2.RP-299~~ ~~IV.B2.RP-300~~ ~~IV.B2.RP-345~~ ~~IV.B2.RP-354~~ ~~IV.B2.RP-386~~ ~~IV.B2.RP-388~~	~~N/A~~ ~~IV.B2-1(R-124)~~ ~~IV.B2-6(R-128)~~ ~~IV.B2-6(R-128)~~ ~~IV.B2-9(R-122)~~ ~~IV.B2-14(R-137)~~ ~~IV.B2-17(R-135)~~ ~~IV.B2-18(R-132)~~ ~~IV.B2-21(R-140)~~ ~~IV.B2-21(R-140)~~ ~~IV.B2-22(R-141)~~ ~~N/A~~ ~~N/A~~ ~~IV.B2-34(R-115)~~ ~~IV.B2-33(R-108)~~ ~~N/A~~ ~~N/A~~ ~~N/A~~ ~~N/A~~
56a	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; or changes in dimensions due to void swelling or distortion; or loss of preload due to thermal	Chapter XI.M16A, "PWR Vessel Internals"	No	IV.B3.RP-315 IV.B3.RP-318 IV.B3.RP-359 IV.B3.RP-360 IV.B3.RP-362 IV.B3.RP-364 IV.B3.RP-366 IV.B3.RP-365 IV.B3.RP-326	IV.B3-7 (R-165) IV.B3-8 (R-163) N/A N/A N/A N/A N/A N/A N/A

Table 3.1-1 Summary of Aging Management Programs for Reactor Vessel, Internals, and Reactor Coolant System Evaluated in Chapter IV of the GALL Report							
ID	**Type**	**Component**	**Aging Effect/Mechanism**	**Aging Management Programs**	**Further Evaluation Recommended**	**Rev2 Item**	**Rev1 Item**
			and irradiation enhanced stress relaxation or creep; or loss of material due to wear				
56b	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; 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	Chapter XI.M16A, "PWR Vessel Internals"	No	IV.B3.RP-317 IV.B3.RP-331 IV.B3.RP-359a IV.B3.RP-361 IV.B3.RP-362b	IV.B3-7 (R-165) N/A N/A N/A N/A
56c	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; 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	Chapter XI.M16A, "PWR Vessel Internals"	No	IV.B3.RP-319 IV.B3.RP-332 IV.B3.RP-334a IV.B3.RP-336 IV.B3.RP-357	IV.B3-9 (R-162) IV.B3-17 (R-156) N/A IV.B3-22 (R-170) N/A
58a	PWR	Stainless steel (SS, including CASS, PH SS or martensitic SS) or 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	Chapter XI.M16A, "PWR Vessel Internals"	No	IV.B4.RP-240 IV.B4.RP-240a IV.B4.RP-242 IV.B4.RP-247b IV.B4.RP-248b IV.B4.RP-249 IV.B4.RP-251 IV.B4.RP-251a IV.B4.RP-252 IV.B4.RP-254b IV.B4.RP-256b IV.B4.RP-258 IV.B4.RP-259 IV.B4.RP-401	IV.B4-1 (R-128) N/A IV.B4-4 (R-183) N/A N/A IV.B4-12 (R-196) IV.B4-15 (R-190) N/A IV.B4-16 (R-188) N/A N/A IV.B4-4 (R-183) IV.B4-31 (R-205) N/A

Table 3.1-1 Summary of Aging Management Programs for Reactor Vessel, Internals, and Reactor Coolant System Evaluated in Chapter IV of the GALL Report							
ID	**Type**	**Component**	**Aging Effect/Mechanism**	**Aging Management Programs**	**Further Evaluation Recommended**	**Rev2 Item**	**Rev1 Item**
58b	PWR	Stainless steel (SS, including CASS, PH SS or martensitic SS) or 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	Chapter XI.M16A, "PWR Vessel Internals"	No	IV.B4.RP-245b IV.B4.RP-246b IV.B4.RP-254b IV.B4.RP-260 IV.B4.RP-243 IV.B4.RP-243a IV.B4.RP-250 IV.B4.RP-375a	N/A N/A N/A IV.B4-31 (R-205) IV.B4-1 (R-128) N/A IV.B4-12 (R-196) N/A
59a	PWR	Stainless steel (SS, including CASS, PH SS or martensitic SS) or nickel alloy Westinghouse 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 thermal and irradiation enhanced stress relaxation or creep; or loss of material due to wear	Chapter XI.M16A, "PWR Vessel Internals"	No	IV.B2.RP-270 IV.B2.RP-272 IV.B2.RP-296 IV.B2.RP-297 IV.B2.RP-302a IV.B2.RP-354 IV.B2.RP-388 IV.B2.RP-300	IV.B2-1 (R-124) IV.B2-6 (R-128) N/A N/A N/A N/A N/A N/A
59b	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; 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	Chapter XI.M16A, "PWR Vessel Internals"	No	IV.B2.RP-274 IV.B2.RP-278a 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-388a	IV.B2-6 (R-128) N/A IV.B2-17 (R-135) IV.B2-21 (R-140) N/A N/A IV.B2-21 (R-140) IV.B2-22 (R-141) N/A
59c	PWR	Stainless steel (SS, including CASS, PH SS or martensitic SS) or nickel alloy Westinghouse reactor	Loss of fracture toughness due to neutron irradiation embrittlement and for CASS, martensitic SS, and PH SS	Chapter XI.M16A, "PWR Vessel Internals"	No	IV.B2.RP-285 IV.B2.RP-288 IV.B2.RP-299 IV.B2.RP-356	IV.B2-14 (R-137) IV.B2-18 (R-132) IV.B2-34 (R-115) N/A

Table 3.1-1 Summary of Aging Management Programs for Reactor Vessel, Internals, and Reactor Coolant System Evaluated in Chapter IV of the GALL Report							
ID	Type	Component	Aging Effect/Mechanism	Aging Management Programs	Further Evaluation Recommended	Rev2 Item	Rev1 Item
		internal "Existing Programs" components exposed to reactor coolant and neutron flux	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				

Appendix C

STAFF RESPONSE TO PUBLIC COMMENTS ON DRAFT LICENSE RENEWAL INTERIM STAFF GUIDANCE 2011-04

Source of Comments
I. Comments from Jean Smith, Electric Power Research Institute Materials Reliability Program (EPRI-MRP) and the Pressurized Water Reactor Owners Group Materials Subcommittee (PWROG-MSC) (ADAMS Accession No. ML12146A267)
II. Comments from Mark Richter, Nuclear Energy Institute (NEI) (ADAMS Accession No. ML12144A147)

#	Source ID	Summary of Comment	Response
1	I-1	The NRC reviewed and approved with limitations MRP-227 Revision 0, and subsequently, MRP-227-A was published to incorporate the SER additions. All needed actions for licensees are contained in MRP-227-A. As a result, it is appropriate for the NRC to review a licensee's PWR reactor internals aging management program against the criteria contained in MRP-227-A. As such, it is not necessary to include all the details currently in NUREG-1800 and NUREG-1801 regarding PWR reactor internals, and instead, only a reference to MRP-227-A should be made. Outlining the requirements for reactor internals in the Interim Staff Guidance may lead to confusion with respect to the implementation of duplicate requirements, may cause undue NRC staff burden reconciling the documents each time MRP-227 is revised by the industry, and will likely lead to human errors in document alignment through future revisions.	The staff agrees with the comment, in part, that it is not necessary to have the level of detail included in LR-ISG-2011-04 issued for public comment regarding PWR reactor vessel internal (RVI) components. However, the staff does not agree that the final LR-ISG-2011-04 should only reference MRP-227-A; instead reference to the topical report should be made only when it is appropriate. Revisions were made to eliminate duplication of information for RVIs that is detailed in MRP-227-A. The following is a summary of the revisions that have been incorporated into final LR-ISG-2011-04 as a result of this comment: *Revision to GALL Report Aging Management Program (AMP) XI.M16A* In general, GALL Report AMP XI.M16A, "PWR Vessel Internals," in final LR-ISG-2011-04 references MRP-227-A in the program elements and does not delineate the MRP-227-A inspection and evaluation guidelines for PWR RVIs. In addition, areas resolved in the staff's safety evaluation (SE), Revision 1, for MRP-227 and Applicant/Licensee Action Items (A/LAI) are not addressed in GALL Report AMP XI.M16A in final LR-ISG-2011-04. *Revision to SRP-LR Table 3.1-1* Final LR-ISG-2011-04 does not incorporate specific reference to "Primary Category," "Expansion Category," or "Existing Program" inspection and evaluation guidelines into the "Rev. 2 Item" column in the aging management review (AMR) line items for PWR RVIs in SRP-LR Table 3.1-1. In addition, the "Component" column for PWR RVIs in SRP-LR Table 3.1-1 in final LR-ISG-2011-04 is based on the commodity groups and inspection categories in MRP-227-A. *Revision to GALL Tables IV.B2, IV.B3, and IV.B4* GALL Tables IV.B2, IV.B3, and IV.B4 in final LR-ISG-2011-04 do not reference inspection categories and MRP-227-A inspection and evaluation guidelines. *Revision to SRP-LR Further Evaluation Recommendations for PWR RVIs* Areas resolved in the staff's SE, Revision 1, for MRP-227 and A/LAIs are not addressed in the Further Evaluation sections of the SRP-LR in final LR-ISG-2011-04 (i.e., these SRP-LR sections were deleted and do appear in final LR-ISG-2011-04). In addition, final LR-ISG-2011-04 recommends that license renewal applicants for PWRs provide their responses to the

#	Source ID	Summary of Comment	Response
			A/LAIs for MRP-227-A in Appendix C of the LRA.
2	I-2	Commenter referenced statement in Section 3.1.2.2.9.A.1 of Appendix A of LR-ISG 2011-04. This statement requires that licensees include responses to applicant action items in both Appendix C of the LRA and in appropriate further evaluation sections of the LRA. This duplication of information provides no significant value to the reviewers. It is recommended that all A/LAI responses be included only in Appendix C, so they are in an easily-referenced location. Any additional discussion of the A/LAIs in the further evaluation sections of the SRP should be limited to identifying each of the items requiring responses and any details necessary to ensure responses are adequate. Any other items requiring discussion of the A/LAI responses in further evaluation sections of the LRA should be deleted or reference made to Appendix C of the LRA.	The staff agrees with the comment that the responses to A/LAIs are to be provided in Appendix C of the LRAs. Thus, final LR-ISG-2011-04 recommends that license renewal applicants for PWRs provide their responses to the A/LAIs for MRP-227-A in Appendix C of the LRA. In addition, as a result of the staff's resolution of Source ID I-1, areas resolved in the staff's SE, Revision 1, for MRP-227 and A/LAIs are not addressed in the Further Evaluation sections of the SRP-LR in final LR-ISG-2011-04. As a result of the staff's resolution of Source ID I-1, final LR-ISG-2011-04 does not incorporate SRP-LR Section 3.1.2.2.9.A.1.
3	I-3	In NUREG-1801 Revision 2 XI.M16A Program Description, last paragraph, as well as in ISG-LR-2011-04 Section 3.1.2.2.9.A.2, both an aging management program and an inspection plan are required to be submitted as part of an applicant's license renewal application. However nowhere in these two documents is there any clear guidance on the information that should be included in an inspection plan. This ambiguity could lead to applicants submitting information that might not meet NRC needs in this area. In order to address this situation it is requested that the aging management program and inspection plan for an applicant be clearly defined. It is proposed that the aging management program address the 10 program element recommendations for PWR RVI components in GALL AMP XI.M16A, PWR Vessel Internals (AMP XI.M16A in NUREG-1801, Revision 2). The inspection plan could be included within a program (i.e. a program/plan) or be a separate document if submitted with a license renewal application. The industry believes these elements are satisfied by the applicable line items from Tables 4-1 through 4-9 and Tables 5-1 through 5-3 of MRP-227-A. The inspection plan submitted as part of a license renewal application (LRA) should be included in Appendix C of the LRA along with the responses to the A/LAI items since it is a requirement of A/LAI No. 8.	The staff agrees, in part, with the comment in that better guidance regarding the inspection plan is needed to avoid confusion. Regulatory Issue Summary (RIS) 2011-07, "License Renewal Submittal Information For Pressurized Water Reactor Internals Aging Management," dated July 21, 2011, provides the staff's expectations for Category D plants (PWR plant licensees that had not submitted their LRAs but plan to submit an LRA in the future) to submit, for NRC staff review and approval, an AMP for vessel internals that is consistent with MRP-227-A. An "inspection plan" is one aspect of satisfying A/LAI No. 8 of the staff's SE, Revision 1, for MRP-227. An "inspection plan" provides information about the RVI components to be inspected and a description of how they will be managed for age-related degradation (e.g., examination method, frequency, acceptance criteria, coverage, etc.). The staff expects that the details of an "inspection plan" for Category D plants will be incorporated into the LRA submittal as part of the 10-element AMP and AMR line items. Thus, consistent with RIS 2011-07, the staff does not expect Category D plants to provide a separate document that contains an "inspection plan" in response to A/LAI No. 8. In order to avoid duplication and confusion, as part of the resolution to Source ID I-1, areas resolved in the staff's SE, Revision 1, for MRP-227 and A/LAIs are not addressed in the Further Evaluation sections of the SRP-LR in final LR-ISG-2011-04. In addition, final LR-ISG-2011-04 recommends that license renewal applicants for PWRs provide their

#	Source ID	Summary of Comment	Response
			responses to the A/LAIs for MRP-227-A in Appendix C of the LRA. In doing this, the explicit reference to an "inspection plan" is avoided in the body of the AMP, and "inspection plan" is only referenced as part of A/LAI No. 8 in the staff's SE, Revision 1, for MRP-227. However, the staff does not agree with the Commenter's general claim with respect to what satisfies an inspection plan per A/LAI No. 8, as additional guidance is outlined in the SE, Revision 1, for MRP-227, and fulfillment of that action item will depend on each applicant's plant-specific review.
4	I-4	The stipulation of appropriate inspection methodologies for these reactor internals components has already been addressed in the review of MRP-227-A. The recommended inspection methods have already been reviewed and found to be adequate to detect the relevant conditions. The AMP attribute that is at issue is not detection of aging effects; instead, the issue is the applicant's corrective action program, and the disposition of relevant conditions through supplemental examination or engineering evaluation, both of which are outside the scope of the Mandatory or Needed requirements of MRP-227-A. Standards for engineering evaluation are addressed in Section 6 of MRP-227-A and in the methodologies described in WCAP-17096. These recommendations are based on the practice used in Section XI of the ASME code and are consistent with existing aging management programs. Further justification for the use of the VT-3 examination is not necessary and should not be required by the ISG. It is recommended that Acceptance Criteria Item 3.1.2.2.9.A.7 (Use of VT-3 Visual Inspection Techniques for Detection of Cracking) be completely eliminated and replaced by a limited requirement to address the acceptability of VT-3 as a management approach for components that 1) were not already considered for aging management in the development of MRP-227-A, 2) are evaluated to require active aging monitoring, and 3) are non-redundant. The Commenter provided justification for its recommendation.	The staff agrees with the comment, in part, that final LR-ISG-2011-04 address the acceptability of VT-3 as a management approach for certain components. Thus, final LR-ISG-2011-04 does not incorporate SRP-LR Sections 3.1.2.2.9.A.7, 3.1.2.2.9.C.1, and 3.1.2.2.9.C.4. However, the staff's position on the use of VT-3 to detect cracking will continue to be documented in the "Detection of Aging Effects" program element in GALL Report AMP XI.M16A, which states, in part, the following: "...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."
5	I-5	Appendix A – Section 2, Acceptance Criteria Item 3.1.2.2.9.A.9 (Identification of TLAAs for PWR-Design RVI Components) on Page A-20 and A-21 stipulates that, "in order to satisfy the requirements of the ASME Code, Section III, Subsections NG-2160 and NG-3121, license renewal applicants demonstrating acceptability of RVI components with design-basis cumulative usage factor (CUF) analyses that are TLAAs should include the effects of the reactor	The staff agrees with the comment, in part, that the evaluation of environmental effects for PWR RVI core support structures should not be incorporated in SRP-LR Section 3.1.2.2.9.A.9 in final LR-ISG-2011-04. However, the staff does not agree with the commenter's statement that the evaluation of time-limited aging analyses for the reactor internals should be addressed in accordance with the existing 10 CFR Part 54 requirements without the need to include environmental effects.

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		coolant system water environment in the fatigue CUF analyses." The Commenter provided its justification for removal of this last sentence.	As a result of the staff's resolution of Source ID I-1, areas resolved in the staff's SE, Revision 1, for MRP-227 and A/LAIs are not addressed in the Further Evaluation sections of the SRP-LR in final LR-ISG-2011-04. Final LR-ISG-2011-04 recommends that license renewal applicants for PWRs provide their responses to the A/LAIs for MRP-227-A in Appendix C of the LRA. As a result of the staff's resolution of Source ID I-1, final LR-ISG-2011-04 does not incorporate SRP-LR Section 3.1.2.2.9.A.9. To the extent that the commenter does not agree with the need to address evaluation of environmental effects, the staff's SE, Revision 1 for MRP-227-A documents the basis for limitations and conditions being placed on the use of MRP-227 as well as A/LAIs that shall be addressed by applicants/licensees who choose to implement the NRC-approved version of MRP-227. Specifically, the topic of environmentally-assisted fatigue for PWR RVIs is addressed in A/LAI No. 8, Item 5 of MRP-227-A. Thus, the intent of LR-ISG-2011-04 is not to supplement or modify the evaluation in the staff's SE, Revision 1.
6	I-6	The component-specific AMR items described in Appendix-A, Sections 4, 5 and 6 are based on migration from NUREG-1801. As a result the listing is more complex than the approved MRP-227-A tables. For example, there are approximately 25 items in Section 5 that classify as "Primary" component examinations, whereas the equivalent component list in MRP-227-A contains only 13 items. The component content is very similar but the breakdown is complex. A key advantage of aligning license renewal commitments to the MRP-227-A format is to facilitate important, industry-wide program updates based on Operating Experience through the NEI 03-08 process. The alignment between MRP-227-A and NUREG-1801 is compromised by embedding item detail in the ISG format. It is recommended that NUREG-1801 refer existing AMR items to "the applicable MRP-227-A table" and retain detail only for those items which may be beyond the scope of MRP-227-A. This will significantly reduce applicant and NRC staff burden, and improve integration of evolutionary changes through the NEI 03-08 process.	The staff does not agree with the comment recommending that NUREG-1801 refer existing AMR line items to "the applicable MRP-227-A table" and retain detail only for those items which may be beyond the scope of MRP-227-A. In accordance with 10 CFR 54.21(a)(3) for each structure and component identified as part of the integrated plant assessment (IPA), the LRA is to demonstrate that the effects of aging will be adequately managed so that the intended function(s) will be maintained consistent with the current licensing basis for the period of extended operation. The IPA is independent of the line items in MRP-227-A and the GALL Report and may also result in additional components beyond the generic lists in these documents. This requires that the LRA provide a complete listing of AMR line items, which may include items consistent with MRP-227-A and the GALL Report and may also result in additional components beyond the generic lists in these documents. Thus, the number and content of AMR line items in the inspection tables of MRP-227-A are not the only basis for determining the AMR line items in the GALL Report. Similarly, the AMR line items in the GALL Report are not the only basis for determining the aging effects requiring management for components or establishing the AMR line items that are included in an LRA. However, final LR-ISG-2011-04 incorporates revisions to SRP-LR Table 3.1-1 and GALL Tables IV.B2, IV.B3, and IV.B4 as summarized in the

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			staff's resolution to Source ID I-1.
7	I-7	ISG implementation of Applicant/Licensee Action items from the MRP-227-A SER is by way of notes to AMR items listed in Sections 4, 5 and 6. This could be addressed by reference to the appropriate SER action items. It is recommended that the required evaluations would be documented in a single location specified by the ISG rather than associated with individual items. Associating these actions with each individual AMR item increases the burden for both the applicant and NRC staff reviewer.	The staff agrees with the comment that associating A/LAIs with each individual AMR line item increases the burden for both the applicant and NRC staff reviewer. As part of the resolution to Source ID I-1, final LR-ISG-2011-04 incorporates revisions to SRP-LR Table 3.1-1 and GALL Tables IV.B2, IV.B3, and IV.B4. Specifically, GALL Tables IV.B2, IV.B3, and IV.B4 were revised to be consistent with the format of AMR items in the GALL Report for non-RVI components and the footnotes in the "Further Evaluation" column of these tables were deleted.
8	I-8	The draft ISG requires Applicants to develop and submit evaluation of inaccessible Reactor Vessel Internal components in accordance with Note 3 to Sections 4 and 5, and Note 2 to Section 6. With the exception of A/LAI #6 of the MRP-227-A SER, these evaluations have been addressed during review and approval of the Industry program. The requirement to develop, submit and review the inspection basis is unnecessary. It is recommended that this note be eliminated.	The staff agrees with the comment that it is not necessary to provide an evaluation of inaccessible RVI components, with the exception of A/LAI No. 6 of MRP-227-A. As part of the resolution to Source ID I-7, final LR-ISG-2011-04 incorporates revisions to delete the further evaluation footnotes from GALL Tables IV.B2, IV.B3, and IV.B4. As a result of staff's resolution to Source ID I-1, areas resolved in the staff's SE, Revision 1, for MRP-227 and A/LAIs are not redundantly addressed in the Further Evaluation sections of the SRP-LR in final LR-ISG-2011-04.
9	I-9	MRP-227-A provides applicants with an alternative to the defined inspection requirements when plant-specific analyses of accumulated fatigue usage are performed. Applicants may choose to either inspect in accordance with the approved MRP-227-A schedules, or perform analyses. In cases where Applicants perform analyses to relax MRP-227-A requirements, those analyses would be submitted for NRC staff approval in accordance with A/LAI 8. The ISG is unclear regarding these alternatives. For example item IV.B3.RP-343 appears to require physical examinations to support acceptance of the TLAA. The industry recommends that the ISG refer to MRP-227-A and the associated A/LAI requirement discussions.	The staff agrees with the comment that LR-ISG-2011-04 refer to MRP-227-A and the associated A/LAI discussions for alternatives or deviations to the inspection and evaluation guidelines in MRP-227-A. It is the responsibility of the license renewal applicant to demonstrate in accordance with 10 CFR 54.21(a)(3) that it can adequately manage aging of RVIs for the period of extended operation, whether through the use of MRP-227-A or alternatives. If a TLAA exists for a RVI, in accordance with 10 CFR 54.21(c)(1)(iii), an applicant may choose to demonstrate the effects of aging on the intended function of the component will be adequately managed for the period of extended operation. It is incumbent on the license renewal applicant to provide this demonstration of aging management, which can include the use of MRP-227-A or an appropriate alternative. In order to avoid redundancy, areas resolved in the staff's SE, Revision 1, for MRP-227 and A/LAIs are not addressed again in the Further Evaluation sections of the SRP-LR in final LR-ISG-2011-04. In addition, final LR-ISG-2011-04 recommends that license renewal applicants for PWRs provide their responses to the A/LAIs for MRP-227-A in Appendix C of the LRA.

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10	I-10	Item 9.B.1 of the ISG notes that Section 3.2.5.3 of the NRC SE (Revision 1) on MRP-227 Revision 0 recommends that the applicant consider replacement or inspection activities with regard to the Control Rod Guide Tube (CRGT) split pins if the pins are currently fabricated with Alloy X-750 or Type 316 stainless steel material. A review of the referenced section of the SE does not reach the conclusion that this specificity of action is required; the SE requirement is to evaluate the adequacy of the plant-specific existing program to ensure that the aging degradation is adequately managed during the extended period of operation. The SE direction is on evaluation of the performance of the existing program and does not suggest that it should be changed to include inspections. Therefore the industry considers the specificity of direction provided in the SE to be sufficient and the ISG should not provide alternate direction.	The staff agrees with the comment that there is an inconsistency between SRP-LR Section 3.1.2.2.9.B.1 in draft LR-ISG-2011-04 and Section 3.2.5.3 of the staff's SE, Revision 1, for MRP-227. As part of the staff's resolution to Source ID I-1, areas resolved in the staff's SE, Revision 1, for MRP-227 and A/LAIs are not addressed in the Further Evaluation sections of the SRP-LR in final LR-ISG-2011-04. In addition, final LR-ISG-2011-04 recommends that license renewal applicants for PWRs provide their responses to the A/LAIs for MRP-227-A in Appendix C of the LRA. As a result of the staff's resolution of Source ID I-1, final LR-ISG-2011-04 does not incorporate SRP-LR Section 3.1.2.2.9.B.1. Specific to the Westinghouse CRGT split pins, A/LAI No. 3 recommends an evaluation to consider the need to replace the Alloy X-750 split pins, if applicable, or an inspection of the replacement type 316 stainless steel split pins to ensure that cracking has been mitigated and that aging degradation is adequately monitored during the extended period of operation. Thus, the intent of LR-ISG-2011-04 is not to supplement or modify the evaluation in the staff's SE, Revision 1, but rather, to recommend that the response to A/LAI No. 3 of MRP-227-A be appropriately documented in Appendix C of the LRA.
11	I-11	Section C.3, page A23 of LR-ISG 2011-04 states that per MRP-227-A, "...EVT-1 inspections of certain CE-design components would be necessary if the design basis fatigue TLAAs for the components could not demonstrate that fatigue-induced cracking would be adequately managed..." This statement does not accurately represent MRP-227-A Table 4-2, because it assumes that the fatigue evaluations required by the MRP-227-A table item already exist and are part of the current licensing basis, and therefore are formally classifiable as TLAAs. In fact, many, if not all, of the older CE design reactor internals were not qualified to the fatigue rules of ASME III, so TLAAs as defined in 10 CFR Part 54 do not exist. Further, page A24 of the draft ISG states "Otherwise, CE-design applicants for renewal are requested to credit the MRP's EVT-1 basis in MRP-227-A as the applicable aging management basis if either: (1) the CLB does not include applicable CUF or It fatigue analyses for these components;…" This statement appears to compel the applicant who does not have a current licensing basis TLAA to perform EVT-1 inspections. MRP-227-A clearly does not require inspections based solely on the lack of a current licensing basis TLAA. In fact, it only requires that a fatigue evaluation be	The staff agrees with the comment that the discussion related to CE-designed lower core flange welds, core support plates, and fuel alignment plates in SRP-LR Section 3.1.2.2.9.C.3 in draft LR-ISG-2011-04 is not clear. As a result of the staff's resolution of Source ID I-1, areas resolved in the staff's SE, Revision 1, for MRP-227 and A/LAIs are not addressed in the Further Evaluation sections of the SRP-LR in final LR-ISG-2011-04. As a result of these revisions, final LR-ISG-2011-04 does not incorporate SRP-LR Section 3.1.2.2.9.C.3. 10 CFR 54.21(a)(1) requires that license renewal application contain an IPA that must, for those systems, structures, and components within the scope of Part 54, identify and list those structures and components subject to an aging management review (AMR). The components evaluated in MRP-227-A may not fully encompass the components identified in an IPA, as required by 10 CFR 54.21(a)(1), and therefore, should not be considered a substitute for performance of an IPA. The aging effects requiring management for RVIs are not governed by

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		performed to determine if a fatigue issue might exist; and if so, where would inspection be focused to manage it. The method of the fatigue evaluation was intended to be the usual engineering practice, for example by comparison of the number expected operating transient cycles to those specified by design, or by stress analysis if required.	MRP-227-A; rather the content of MRP-227-A serves to assist a PWR license renewal applicant. In accordance with 10 CFR 54.21(a)(3), the effects of aging are to be managed for all applicable aging effects for a particular component, which may be broader than the aging effects identified in MRP-227-A and the GALL Report for RVIs. It is the responsibility of the license renewal applicant to demonstrate that it can adequately manage aging of RVIs for the period of extended operation, whether through the use of MRP-227-A or alternatives. Therefore, if the CE-designed lower core flange welds, core support plates, and fuel alignment plates are subject to an AMR and fatigue is an applicable aging effect, regardless if there is a TLAA, the LRA must demonstrate that fatigue will be adequately managed in accordance with 10 CFR 54.21(a)(3).
12	I-12	For A/LAI No. 2, when comparing the licensee renewal AMR from BAW-2248A to the tables in MRP-189, the locking devices for the vent valve were identified as a possible "Primary" component. The original vent valves located next to outlet nozzles failed due to flow induced vibration, and those valves next to the nozzles were replaced with locking devices made containing Alloy 600. It is recommended that Table IV Reactor Vessel, Internals, and Reactor Coolant System, B4 Reactor Vessel Internals (PWR) – Babcock and Wilcox on page A-124 of LR-ISG 2011-04 be revised to include a line item addressing Alloy 600 replacement vent valve locking devices, which are subject to aging degradation due to primary water stress corrosion cracking (PWSCC).	The staff agrees with the comment to include an AMR line item for cracking of B&W vent valve locking devices made from Alloy 600 materials in GALL Table IV.B4 of draft LR-ISG-2011-04. Final LR-ISG-2011-04 incorporates the core support shield vent valve top and bottom retaining rings to be managed for cracking in GALL AMR Item IV.B4.RP-252a.
13	I-13	In Item 8 on page A-11 of the LR-ISG, the second sentence appears to be incomplete with respect to the statement pertaining to "…confirming that the quality of inspections, flaw evaluations, and corrective actions performed under this program." It is recommended that the revised statements be reviewed for completeness.	The staff agrees with the comment that the sentence in the "Confirmation Process" program element in GALL Report AMP XI.M16A of draft LR-ISG-2011-04 is incomplete. Final LR-ISG-2011-04 completes this sentence in the "Confirmation Process" program element.
14	I-14	Item 3 on page A-16 of the LR-ISG should reference NRC SE Section 3.2.5.1 and not Section 3.5.1. It is recommended that this reference be revised.	The staff agrees with the comment that SRP-LR Section 3.1.2.2.9.A.3 in draft LR-ISG 2011-04 should reference NRC SE Section 3.2.5.1 and not Section 3.5.1. As a result of the staff's resolution of Source ID I-1, areas resolved in the staff's SE, Revision 1, for MRP-227 and A/LAIs are not addressed in the Further Evaluation sections of the SRP-LR in final LR-ISG-2011-04. As a

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			result of these revisions, final LR-ISG-2011-04 does not incorporate SRP-LR Section 3.1.2.2.9.A.3.
15	I-15	Item D.1 on page A-25 discusses evaluation "Acceptance Criteria" recommendations applicable to Babcock and Wilcox reactor internals. In general, A/LAI 4 is not specific relative to the wording for the manner in which the items were stress relieved, and it was stated that a "stress relief process" was used. In Item D.1, the wording used in some cases implies a "post-weld heat treatment" process. The words "stress relief process" should be used consistently without the implication of a heat treatment process only. In addition, the requirements in Item D.1 appear to go beyond the requirements of the A/LAI as it was written and approved by the MRP-227-A SER.	The staff agrees with the comment to use the terminology "stress relief process" consistently throughout SRP-LR Section 3.1.2.2.9.D.1 of draft LR-ISG-2011-04. Final LR-ISG-2011-04 does not use the term "post-weld heat treatment" and this term is replaced with the term "stress relief process." In addition, as a result of the staff's resolution of Source ID I-1, areas resolved in the staff's SE, Revision 1, for MRP-227 and A/LAIs are not addressed in the Further Evaluation sections of the SRP-LR in final LR-ISG-2011-04. As a result of these revisions, final LR-ISG-2011-04 does not incorporate SRP-LR Section 3.1.2.2.9.D.1.
16	II-1	Page A-51 – IV.B2.RP-300 Alignment and interfacing components, such as hold down springs, are addressed in MRP-227-A. Based on MRP-227-A, the intent of the GALL was only to apply to hold down springs made from Type 304 Stainless Steel (SS). The possibility of thermal embrittlement of hold down springs made from Type 403 martensitic SS is not addressed. The issue is, however, discussed in the proposed SRP section 3.1.2.2.9.A.6 and in applicant action item 7 of the SER (Revision1). Proposed Change: Include the words "applicable to hold down springs fabricated from Type 304 SS" and add a line item to address thermal embrittlement for hold down springs fabricated for Type 403 stainless steel.	The staff agrees with the comment that the possibility of thermal embrittlement for Type 403 martensitic stainless steel hold down springs is not addressed. Final LR-ISG-2011-04 does not use the term "(Aust. SS Material)" in the "Material" column in GALL AMR Item IV.B2.RP-300. Furthermore, the use of the term "Stainless Steel" in GALL AMR Item IV.B2.RP-300 is generic and includes all grades of "stainless steel" as defined in GALL Table IX.C, "Selected Definitions & Use of Terms for Describing and Standardizing – MATERIALS." With these revisions hold down springs made from Type 403 martensitic SS are addressed in GALL AMR Item IV.B2.RP-300.
17	II-2	Page A-16 – Section 3.1.2.2.9.A.3, second paragraph There is little guidance on Applicant Action Item #2 related to additional RVI piece parts and what was used during the development of MRP 191. Utilities are left to draw a conclusion that unless the utility implemented a modification beyond the vendor's recommendation, all of the piece parts in the reactor vessel were considered during the development of MRP-189, 191 and 227-A. Proposed Change: Add verbiage to provide additional guidance to allow utilities to make the assumption that unless a utility implemented modifications beyond that recommended by the	The staff does not agree with the comment, in particular the inference that, unless a utility implemented modifications beyond that recommended by the vendor of the RVI, all of the piece parts of the RVI were considered during the development of MRP-189, 191 and 227-A. The methodology and results of a topical report, such as MRP-227-A, cannot be assumed to be generically bounding for every plant. The IPA described in the response to Source ID I-11 is a plant-specific evaluation performed by a license renewal applicant. Thus, the components evaluated in MRP-189, 191 and 227-A may not fully encompass the components identified in an applicant's IPA and therefore, should not be considered a substitute for performance of an IPA. The

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		vendor of the RVI, then all of the piece parts of the RVI were considered during the development of MRP-189, 191 and 227-A.	aging effects requiring management for RVIs may be broader than the aging effects identified in MRP-189, 191 or 227-A. It is the responsibility of the license renewal applicant to demonstrate, in accordance with 10 CFR 54.21(a)(3), that it can adequately manage aging of RVIs for the period of extended operation, whether through the use of MRP-227-A or alternatives. The content in MRP-189, 191 or 227-A only serves to assist a PWR license renewal applicant. However, as addressed in the staff's resolution to Source ID I-1, in order to avoid redundancy, areas resolved in the staff's SE, Revision 1, for MRP-227 and A/LAIs are not addressed in the Further Evaluation sections of the SRP-LR in final LR-ISG-2011-04. In addition, final LR-ISG-2011-04 recommends that license renewal applicants for PWRs provide their responses to the A/LAIs for MRP-227-A in Appendix C of the LRA. As a result of these revisions, final LR-ISG-2011-04 does not incorporate SRP-LR Section 3.1.2.2.9.A.3.
18	II-3	Page A-16 – Section 3.1.2.2.9.A.3 The words in the third paragraph are confusing and it is not clear what is meant by plant specific AMR line items or why Note E would be appropriate. For those applicants whose plant-specific review results in identification of additional components for inspection or different component inspection categories from those identified in MRP-227-A, the applicant is requested to identify the changes in the component inspection categories as either plant-specific AMR line items or NEI Note E consistent with GALL AMR items (whichever is applicable) in their Table 2 AMR line items for their PWR RVI components. Proposed Change: It is suggested that if only a component line item or two that is not in GALL is being added then an exception can be taken to the program and justification be added that includes inspection specifics such as method and acceptance criteria such that the whole program doesn't have to be evaluated as a plant specific program.	The staff agrees with the comment that portions of SRP Section 3.1.2.2.9.A.3 are confusing. As a result of the staff's resolution of Source ID I-1, areas resolved in the staff's SE, Revision 1, for MRP-227 and A/LAIs are not addressed in the Further Evaluation sections of the SRP-LR in final LR-ISG-2011-04. As a result of these revisions, final LR-ISG-2011-04 does not incorporate SRP-LR Section 3.1.2.2.9.A.3.
19	II-4	Page A-5 – last paragraph Page A-15 – Section 3.1.2.2.9.1.2 The document does not provide clear direction as to what goes into an inspection plan.	The staff agrees that draft LR-ISG-2011-04 does not provide clear direction as to what goes into an inspection plan but does not agree with the commenter's proposed change. The staff does not agree with the Commenter's general claim with respect to what satisfies an inspection plan per A/LAI No. 8, as additional guidance is outlined in the SE, Revision 1, for MRP-227, and fulfillment of that action item will depend on each

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		Proposed Change: Add verbiage to allow utilities to better determine what the inspection plan should consist of (e.g., A Westinghouse design plant should provide unit specific information in the Inspection Plan consistent with tables 4-3, 4-6 and 4-9 of MRP-227-A and the A/LAIs).	applicant's plant-specific review. See the staff's resolution to Source ID I-3. RIS 2011-07 provides the staff's expectations for Category D plants (PWR plant licensees that have not submitted their LRAs but plan to submit an LRA in the future) to submit, for NRC staff review and approval, an AMP for vessel internals that is consistent with MRP-227-A. As an "inspection plan" is one aspect of satisfying A/LAI No. 8 of the staff's SE, Revision 1, for MRP-227. An "inspection plan" provides information about the RVI components to be inspected and a description of how they will be managed for age-related degradation (e.g., examination method, frequency, acceptance criteria, coverage, etc.). The staff expects that the details of an "inspection plan" for Category D plants will be incorporated into the LRA submittal as part of the 10-element AMP and AMR line items. Thus, consistent with RIS 2011-07, the staff does not expect Category D plants to provide a separate document that contains an "inspection plan" in response to A/LAI No. 8. To avoid confusion, final LR-ISG-2011-04 avoids explicit reference to an "inspection plan" in the body of the AMP, and "inspection plan" is only referenced as part of A/LAI No. 8 in the staff's SE, Revision 1, for MRP-227.
20	II-5	Page A-34 – Table 3.1-1 Item 27a It is not clear that this line item is only applicable to hold down springs fabricated from Type 304 SS. Proposed Change: Add Type 304 SS hold down springs.	The staff agrees with the comment that Table 3.1-1, Item 27a, of draft LR-ISG-2011-04 does not clearly address Type 304 stainless steel hold down springs. Final LR-ISG-2011-04 does not include this item, but Westinghouse Type 304 stainless steel hold down springs were incorporated into Table 3.1-1, Item 59a, in final LR-ISG-2011-04, which uses the generic terminology "stainless steel."
21	II-6	Page A-31 – Table 3.1-1 Item 3 Under 'Further Evaluation Recommended' column, it is not clear what "It" stands for? Proposed Change: Provide an explanation.	"It" refers to the parameter being calculated for the cyclical loading analyses. In later editions of the ASME Code Section III, these analyses were referred to as cumulative usage factor (CUF) analyses. Thus, the "I_t" parameter is analogous to the CUF parameter required for Class 1 components designed to more recent editions of the ASME Code, Section III. The subscripted "t" was removed in the formatting during the issuance of draft LR-ISG-2011-04 for public comment. As a result of the staff's resolution of Source ID I-1, SRP-LR Table 3.1-1 Item 3 does not incorporate the reference to the "I_t" parameter in final LR-ISG-2011-04.

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22	II-7	Page A-65 – IV.B2.RP-280 There is confusion regarding what comprises the lower core barrel flange weld for Westinghouse designed plants. This component is still listed in MRP-191, and 227-A for Westinghouse designed plants. MRP-227-A indicates it may be the weld between the core barrel and the lower support forging or casting. Proposed Change: Provide an explanation regarding what this component is.	Page 3-11 of MRP-227-A states that "[t]he lower support forging is welded to and supported by the core barrel, which transmits vertical loads to the vessel through the core barrel flange." In addition, Table 5-3 of MRP-227-A provides the "Examination Acceptance Criteria and Expansion Criteria" for the "Core Barrel Assembly - Lower core barrel flange weld." Footnote 2 of Table 5-3 states that "[t]he lower core barrel flange weld may alternatively be designated as the core barrel-to-support plate weld in some Westinghouse plant designs." No revisions were made as a result of this comment.
23	II-8	Page A-17 – Section 3.1.2.2.9.A.4 In the subject paragraph, it appears the NRC wanted an exception not an enhancement: For those component inspections that do not achieve the inspection coverage criteria stated in the NRC SE (Revision 1) on MRP-227, the applicant is requested to take a deviation from the MRP-defined inspection criteria and describe the process and type of evaluation that will be implemented to evaluate the impact of the aging effects on the inaccessible regions of the components. In this case, the applicant is requested to identify this process as an applicable enhancement of the "monitoring and trending" program element of its RVI Program. Proposed Change: Clarify what is expected.	The staff agrees with the comment that the referenced text in SRP-LR Section 3.1.2.2.9.A.4 of draft LR-ISG-2011-04 is not clear. As a result of the staff's resolution of Source ID I-1, areas resolved in the staff's SE, Revision 1, for MRP-227 and A/LAIs are not addressed in the Further Evaluation sections of the SRP-LR in final LR-ISG-2011-04. In addition, final LR-ISG-2011-04 recommends that license renewal applicants for PWRs provide their responses to the A/LAIs for MRP-227-A in Appendix C of the LRA. As a result of these revisions, final LR-ISG-2011-04 does not incorporate SRP-LR Section 3.1.2.2.9.A.4.
24	II-9	Page A-65 – IV.B2.RP-280 It is not clear how Note 3 in the "Further Evaluation" column is applicable to this GALL Line Item. Proposed Change: Clarify the applicability.	The staff agrees with the comment that the applicability of Note 3 to GALL AMR Item IV.B2.RP-280 is not clear. As a result of the staff's resolution of Source ID I-1, Final LR-ISG-2011-04 incorporates revisions to GALL Tables IV.B2, IV.B3, and IV.B4 as summarized above.
25	II-10	Page 3 In the last paragraph of the Discussion section only table 3-1 is listed for justification of TE for the materials. Tables 3-2 and 3-3 should be mentioned since 3-1 is only for B&W internals. Proposed Change: Add Tables 3-2 and 3-3.	The staff agrees with the comment that Tables 3-2 and 3-3 should be referenced in the last paragraph of the "Discussion" section. However, the "Discussion" section of final LR-ISG-2011-04 no longer references Table 3-1 in MRP-227-A.
26	II-11	Page A-7	The staff agrees with the comment to change the terminology to "Aging Management Requirement" tables in the "Parameters Monitored/Inspected"

#	Source ID	Summary of Comment	Response
		The second paragraph in this Section refers to condition monitoring tables in MRP-227-A. There are no tables with this title in MRP-227-A Proposed Change: Change to Aging Management Requirement tables.	program element. The "Parameters Monitored/Inspected" program element in final LR-ISG-2011-04 states the following: "Specifically, the program implements the parameters monitored/inspected criteria consistent with the applicable tables in Section 4, 'Aging Management Requirement,' in MRP-227-A…"
27	II-12	Page A-9 Only Table 5-1 is listed for acceptance criteria when MRP-227-A contains three tables, 5-1 thru 5-3 Proposed Change: Change to read " Section 5 and Tables 5-1 thru 5-3 of MRP-227"	The staff agrees with the comment to add references to Table 5-2 and 5-3 of MRP-227-A for the "Acceptance Criteria" program element. The "Acceptance Criteria" program element of GALL Report AMP XI.M16A in final LR-ISG-2011-04 references Table 5-1 through 5-3 of MRP-227-A.
28	II-13	Page A-10 The first paragraph on the page says "The program adopts the acceptance criteria for the physical measurement monitoring methods recommended in MRP-227-A, as qualified in Section 3.3.5 and A/LAI No. 5 in Revision 1 of the NRC SE on MRP-227". Section 3.3.5 of the MRP does not specify acceptance criteria so there is nothing to be adopted. It only requires it be developed as discussed in footnote 3. Proposed Change: Change sentence to read "The program includes acceptance criteria for the physical measurement monitoring methods as recommended in MRP-227-A, Section 3.3.5 and A/LAI No. 5 in Revision 1 of the NRC SE on MRP-227".	The staff agrees with the comment that Section 3.3.5 of MRP-227-A does not specify acceptance criteria for physical measurements. However, as a result of the staff's resolution of Source ID I-1, areas resolved in the staff's SE, Revision 1, for MRP-227 and A/LAIs are not addressed in GALL Report AMP XI.M16A in final LR-ISG-2011-04. The "Acceptance Criteria" program element in final LR-ISG-2011-04 states that, in general, the AMP establishes appropriate acceptance criteria for any physical measurement monitoring methods that are credited for aging management of RVIs.
29	II-14	Page A-12 The following sentence relates to notification criteria: "The evaluation in Section 3.5 of Revision 1 of the SE on MRP-227 provides the staff's basis for endorsing the NEI 03-08 implementation process for these programs. This includes NRC's endorsement of the NEI 03-08 criteria for notifying the NRC of any deviation from the I&E methodology in MRP-227-A and justification of the deviation no later than 45 days after approval by a licensee executive." Proposed Change: Delete this sentence as it already is discussed in element 9 where it is appropriate.	The staff agrees with the comment that the sentence associated to the notification criteria already exists in the "Administrative Controls" program element and does not need to be repeated in the "Operating Experience" program element of GALL Report AMP XI.M16A. The "Operating Experience" program element of GALL Report AMP XI.M16A in final LR-ISG-2011-04 does not incorporate this sentence associated with the notification criteria.

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30	II-15	Page A-8 The justification required for the use of VT-3 to detect cracking over that specified in MRP-227A and approved by the staff in the SE that allows its use without the additional limitations and analyses is not needed. Proposed Change: Eliminate need for additional justification if requirements as specified in SER and MRP are met.	The staff agrees with the comment that additional justification for the use of VT-3 to detect cracking is not needed if requirements specified in the SER and MRP are met. As a result of the staff's resolution of Source ID I-4, final LR-ISG-2011-04 does not incorporate SRP-LR Sections 3.1.2.2.9.A.7, 3.1.2.2.9.C.1, and 3.1.2.2.9.C.4. However, the staff's position on the use of VT-3 for the detection of cracking will continue to be documented in the "Detection of Aging Effects" program element in GALL Report AMP XI.M16A.
31	II-16	Page A-22 – Section 3.1.2.2.9.C.1 The justification required for the use of VT-3 to detect cracking over that specified in MRP-227A and approved by the staff in the SE that allows its use without the additional limitations and analyses is not needed. Proposed Change: Eliminate need for additional justification if requirements as specified in SER and MRP are met.	The staff agrees with the comment that additional justification for the use of VT-3 to detect cracking is not needed if requirements specified in the SER and MRP are met. As a result of the staff's resolutions to Source ID I-4 and ID II-15, final LR-ISG-2011-04 does not incorporate SRP-LR Sections 3.1.2.2.9.A.7, 3.1.2.2.9.C.1, and 3.1.2.2.9.C.4. However, the staff's position on the use of VT-3 for the detection of cracking will continue to be documented in the "Detection of Aging Effects" program element in GALL Report AMP XI.M16A.
32	II-17	Page A-23 – Section 3.1.2.2.9.C.3 The option presented as (3), as an alternative basis for accepting the design basis fatigue analyses in accordance with the TLAA acceptance requirement in 10 CFR 54.21(c)(1)(iii) does not make sense when compared to options 1 and 2 Proposed Change: Add the word "the EVT-1 is used" at the beginning	The staff agrees with the comment that the discussion related to CE-designed lower core flange welds, core support plates, and fuel alignment plates in SRP-LR Section 3.1.2.2.9.C.3 of draft LR-ISG-2011-04 is not clear. As a result of the staff's resolution of Source ID I-1, areas resolved in the staff's SE, Revision 1, for MRP-227 and A/LAIs are not addressed in the Further Evaluation sections of the SRP-LR in final LR-ISG-2011-04. As a result of these revisions, final LR-ISG-2011-04 does not incorporate SRP-LR Section 3.1.2.2.9.C.3. Also see the staff's resolution to Source ID I-11, in which the staff clarified that, if the CE-designed lower core flange welds, core support plates, and fuel alignment plates are subject to an AMR and fatigue is an applicable aging effect, regardless if there is a TLAA, then in accordance with 10 CFR 54.21(a)(3), the LRA must demonstrate that fatigue will be adequately managed.
33	II-18	Page A-25 – Section 3.1.2.2.9.D.1 There is no need for a plant-specific enhancement of the	The staff agrees with the comment that there is not a need for a plant-specific enhancement of the "Preventive Actions" program element discussed in SRP-LR Section 3.1.2.2.9.D.1 of draft LR-ISG-2011-04, which

#	Source ID	Summary of Comment	Response
		"preventative actions" program element for their RVI Program enhancement to be identified if an applicant confirms that the welds were appropriately stress-relieved. An enhancement doesn't seem appropriate since the action has already been taken. Proposed Change: Eliminate the need for an enhancement	is associated with A/LAIs No. 4 of MRP-227-A. As a result of the staff's resolution of Source ID I-1, areas resolved in the staff's SE, Revision 1, for MRP-227 and A/LAIs are not addressed in the Further Evaluation sections of the SRP-LR in final LR-ISG-2011-04. In addition, final LR-ISG-2011-04 recommends that license renewal applicants for PWRs provide their responses to the A/LAIs for MRP-227-A in Appendix C of the LRA. As a result of these revisions, final LR-ISG-2011-04 does not incorporate SRP-LR Section 3.1.2.2.9.D.1.
34	II-19	Page A-30 – Table 3.0-1 There is no need for the words "or to applicable NRC further evaluation "acceptance criteria" recommendations in Section 3.1.2.2 of the SRP-LR (i.e., the latest NRC issued version of NUREG-1800)." Specific acceptance criteria do not need to be part of a SAR description. If it is an enhancement it will already be a commitment. Proposed Change: Delete	The staff agrees with the comment that the further evaluation acceptance criteria do not need to be specified as part of a Safety Analysis Report description. Final LR-ISG-2011-04 does not incorporate this second paragraph in the "Description of Program" column for GALL Report AMP XI.M16A in SRP-LR Table 3.0-1. However, 10 CFR 54.21(d) provides the requirements for a Final Safety Analysis Report supplement and states, in part, that it must contain a summary description of the programs and activities for managing the effects of aging. The specificity of such descriptions will depend on the program proposed by each license renewal applicant.
35	II-20	Page A-51 – Table IV.B2 There is no need for specifying the Examination technique in the Program column. Proposed Change: Delete	The staff agrees with the comment that there is no need for specifying the Examination Technique in the "Aging Management Program" column of GALL Table IV.B2. GALL Tables IV.B2, IV.B3 and IV.B4 in final LR-ISG-2011-04 do not incorporate a summary of the examination techniques from the "Aging Management Program" column.
36	II-21	A-77 – Footnotes For note 6, see comments 15 and 16 above on why no justification for using VT-3 exam is required when it was acceptable in SER for 227. This applies to CE and B&W tables that also contain a similar note. Proposed Change: Delete the note	The staff agrees with the comment that additional justification for the use of VT-3 to detect cracking is not needed if requirements specified in the SER and MRP are met. As a result of the staff's resolutions of Source ID I-4 and ID II-15, final LR-ISG-2011-04 does not incorporate SRP-LR Sections 3.1.2.2.9.A.7, 3.1.2.2.9.C.1, and 3.1.2.2.9.C.4. However, the staff's position on the use of VT-3 for the detection of cracking will continue to be documented in the "Detection of Aging Effects" program element in GALL Report AMP XI.M16A. In addition, as part of the staff's resolution to Source ID I-8, the format of GALL Tables IV.B2, IV.B3, and IV.B4 in final LR-ISG-2011-04 is consistent with AMR items in the GALL Report for non-RVI components and does not incorporate the footnotes in the "Further Evaluation" column of these tables.

#	Source ID	Summary of Comment	Response
37	II-22	Page A-102 – Footnote #1 “In conjunction” is repeated in the second sentence. Proposed Change: Delete second in conjunction	The staff agrees with the comment that “in conjunction with” was an editorial error in Note 1. However, as part of the staff’s resolution to Source ID I-8, the format of GALL Tables IV.B2, IV.B3, and IV.B4 in final LR-ISG-2011-04 is consistent with AMR items in the GALL Report for non-RVI components and does not incorporate the footnotes in the “Further Evaluation” column of these tables. As a result of these revisions, the referenced Note 1 is not incorporated in final LR-ISG-2011-04.
38	II-23	Page A-104 – Footnote #8 4th line “No.2 above, and is so” should be and if so. Proposed Change: Correct	The staff agrees with the comment that there is a typographical error in Note 8 of page A-103. However, as part of the staff’s resolution to Source ID I-8, the format of GALL Tables IV.B2, IV.B3, and IV.B4 in final LR-ISG-2011-04 is consistent with AMR items in the GALL Report for non-RVI components and does not incorporate the footnotes in the “Further Evaluation” column of these tables. As a result of these revisions, the referenced Note 8 is not incorporated in final LR-ISG-2011-04.
39	II-24	Page A-54 – Table IV.B2 Water chemistry is not listed as an AMP, with the aging effect of stress corrosion cracking (SCC) and irradiation-assisted stress corrosion cracking (IASCC) such as in line items IV.B2.RP-270a, 345, 399, 299a. This mainly occurs in new line items and also shows up in Table IV.B3 and IV.B4 Proposed Change: Add XI.M2 as an AMP	The staff agrees with the comment that GALL Report AMP X.M2, “Water Chemistry,” is not listed in GALL Table IV.B2. GALL Table IV.B2, IV.B3 and IV.B4 of final LR-ISG-2011-04 include GALL Report AMP X.M2, “Water Chemistry,” as a recommendation to manage cracking by SCC, PWSCC, or IASCC, or loss of material due to pitting or crevice corrosion of RVIs.
40	II-25	Page A-76 – IV.B2.RP-399 As indicated in Table 4-9 of MRP-227-A and the associated note 2, the clevis insert bolts are inspected for wear. To the extent cracking would be visible in the VT-3 inspection, it would of course be addressed; but, the intent of the inspection is to look for wear. Proposed Change: Eliminate this line as an existing inspection program element, or change the AMP description to note the inspection is for gross effects of cracking	The staff agrees with the comment that Table 4-9 of MRP-227-A did not identify cracking as an aging effect requiring management for Westinghouse-design clevis insert bolts of screws but does not agree with the commenter’s proposed change. Relevant operating experience associated with aging may exist that has not been accounted for in MRP-227-A. AMR item IV.B2.RP-399 for cracking of Westinghouse-design clevis insert bolts and screws was included in LR-ISG-2011-04 based on industry operating experience. Appendix A of MRP-227-A states, in part, that “[f]ailures of Alloy X-750 clevis insert bolts were reported by one Westinghouse-designed plant in 2010” and “[a]lthough the failed clevis insert bolts were not removed for metallurgical examination, it can be surmised that the most likely cause of failure was PWSCC.” No revisions were made as a result of this comment.
41	II-26	Page A-67 – IV.B2.RP-285	The staff agrees with the comment to delete the aging mechanism of loss of fracture toughness from AMR item IV.B2.RP-285. Since the clevis bolts

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		As described in MRP-191, the clevis bolts and inserts are not in a high flux region and irradiation embrittlement is not a significant aging mechanism. As indicated in Table 4-9 of MRP-227-A and the associated note 2, the clevis insert bolts are inspected for wear. Also, Note 5 is applied to the further evaluation column; however, Note 5 refers to reduction of fracture toughness due to thermal embrittlement in stainless steel components, while the material listed for this line is nickel alloy. Proposed Change: Eliminate the aging mechanism of loss of fracture toughness from this line and remove note 5 from the further evaluation column.	and inserts are not in a high flux region, GALL AMR Item IV.B2.RP-285 in final LR-ISG-2011-04 does not incorporate the aging effect of loss of fracture toughness due to neutron irradiation embrittlement. As a result of the staff's resolution of Source ID I-8, the format of GALL Tables IV.B2, IV.B3, and IV.B4 in final LR-ISG-2011-04 is consistent with AMR items in the GALL Report for non-RVI components and deleted the footnotes in the "Further Evaluation" column of these tables.
42	II-27	Page A-63 – IV.B2.RP-345 As indicated in Table 5-1 of MRP-191, cracking of the core barrel flange is a concern for the weld rather than the base metal. Table 4-3 specifically identifies the welds as primary components to be inspected for cracking. While inspections of the welds would identify cracking in the adjacent base metal, separately adding cracking as an aging effect to the base metal as an existing component is not consistent with MRP-227-A or existing inspections. Proposed Change: Eliminate base metal cracking as an aging effect in this line.	The staff agrees with the comment to delete base metal cracking from GALL AMR Item IV.B2.RP-345 of draft LR-ISG-2011-04 since MRP-227-A identifies that the adjacent base metal is part of the examination coverage for the "Core Barrel Assembly - Lower core barrel flange weld." Thus, GALL AMR Item IV.B2.RP-345 in final LR-ISG-2011-04 does not reference cracking of the core barrel flange (base metal). GALL AMR IV.B2.RP-345 continues to identify loss of material due to wear for the core barrel flange (base metal).
43	II-28	Page A-9 Flaw evaluation methods include recommendations for flaw depth sizing and for crack growth determinations as well for performing applicable load limit. It should read … "growth determinations as well as for performing." Proposed Change: Change to include missing "as."	The staff agrees with the comment that the sentence is incomplete. This sentence in the "Monitoring and Trending" program element of GALL Report AMP XI.M16A in final LR-ISG-2011-04 is complete.
44	II-29	Page A-49 In the first paragraph under Systems, Structures, and Components thermal shield assembly should be changed to thermal shield or neutron pad assembly to address the newer Westinghouse plants. Also, the component type neutron pad is not addressed in Table B2 or MRP-227.	The staff acknowledges that the component type neutron pad assembly is not addressed in GALL Table IV.B2 or MRP-227-A. However, the intent of this LR-ISG is not to supplement such aspects that are not covered in MRP-227-A. Thus, no revisions were made as a result of this comment. If a PWR license renewal applicant identifies during the IPA that its plant design contains a neutron pad assembly (instead of a thermal shield assembly) and is subject to an AMR, the license renewal applicant must

#	Source ID	Summary of Comment	Response
		Proposed Change: Address recommended change.	identify this assembly in its LRA and propose an adequate means of aging management.
45	II-30	Table IV.B2 The environment "Reactor coolant and neutron flux" is used for all line items/components in Table B2, however not all the components listed in Table B2 will experience a neutron fluence exceeding 10^{17} n/cm2 (E>1MeV) at the end of the period of extended operation. The environment should be more specific based on the location (fluence) of the components. Proposed Change: The Table should note exceptions to the neutron fluence level.	The staff does not agree with the comment to note exceptions with regard to use the term "neutron flux" in GALL AMR items in the GALL Report. The GALL Report generically and conservatively assumes that PWR RVIs are exposed to an environment of "reactor coolant and neutron flux" regardless of the fluence level. The staff anticipates that applicants will address their plant-specific data in their IPA and identify appropriate AMR items. No revisions were made as a result of this comment.
46	II-31	Page A-33 – Table 3.1-1 Item 27 Component was changed to nickel alloy guide tube support pins, however associated Table B2 line items IV.B2.RP-355 and IV.B2.RP-356 were changed to include both nickel alloy and stainless steel. Proposed Change: Clarify	The component in SRP-LR Table 3.1-1, Item 27, which refers to control rod guide tube (CRGT) split pins (support pins), is applicable to both nickel alloy and stainless steel materials. SRP-LR Table 3.1-1 Item 27 in draft LR-ISG-2011-04 was removed and incorporated into Table 3.1-1 Item 53c in final LR-ISG-2011-04.
47	II-32	A-49 – Last paragraph Sentence "EPR MRP methodology left some..." should be changed. Proposed Change: Should read "EPRI MRP methodology left some..."	The staff agrees with the proposed change, however, as a result of the staff's resolution of Source ID I-1 the referenced sentence is not incorporated in final LR-ISG-2011-04.
48	II-33	The following acronyms are used but not included in Appendix B of this ISG; CUF, NRC, SE, and USAR. Proposed Change: Update Appendix B to include all acronyms.	The staff agrees with the comment; however, draft LR-ISG-2011-04 was revised to remove the full list of acronyms in LR-ISG-2011-04, Appendix B. Final LR-ISG-2011-04, Appendix B, was revised to document the mark-up of changes to the GALL Report and SRP-LR. Acronyms in final LR-ISG-2011-04 are defined the first time they are used.
49	II-34	The page numbers for Appendix B are A-165 and A-166, the last page of Appendix A is A-144. Proposed Change: Verify correct pagination.	The staff agrees with the comment and final LR-ISG-2011-04 includes the correct page numbers.

#	Source ID	Summary of Comment	Response
50	II-35	Page A-18 – Section 3.1.2.2.9.A.5 For re-inspection greater than 10 years, further evaluation is redundant and inconsistent with standard GALL AMR and AMP formatting and presentation. Inspection frequencies would be evaluated in AMP element 4 for consistency with MRP-227-A chapter 4 primary, expansion, and existing components inspection tables. If the inspection frequency is identified that is not consistent with MRP-227-A Chapter 4 tables, an exception must be identified and justified. Proposed Change: Delete further evaluation 3.1.2.2.9.A item 5. Item to be addressed by AMP element 4.	The staff agrees with the comment that if an inspection frequency is not consistent with MRP-227-A, an exception must be identified and justified. Furthermore, Section 4.0 of the staff's SE, Revision 1, for MRP-227 provides the "Conditions And Limitations And Applicant/Licensee Plant-Specific Action Items," which specifically states that the re-examination frequency for "Primary" inspection category components shall be on a maximum 10-year interval, unless a plant-specific analysis providing justification for an extended examination frequency is submitted to and approved by the NRC. As a result of the staff's resolution of Source ID I-1, areas resolved in the staff's SE, Revision 1, for MRP-227 and A/LAIs are not addressed in the Further Evaluation sections of the SRP-LR in final LR-ISG-2011-04. In addition, final LR-ISG-2011-04 recommends that license renewal applicants for PWRs provide their responses to the A/LAIs for MRP-227-A in Appendix C of the LRA. As a result of these revisions, final LR-ISG-2011-04 does not incorporate SRP-LR Section 3.1.2.2.9.A.5.
51	II-36	Page A-19 – Section 3.1.2.2.9.A.7 For VT-3 Inspection, further evaluation is redundant and inconsistent with standard GALL AMR and AMP formatting and presentation. VT-3 inspection requirements should be addressed as part of AMP element 3 for consistency with MRP-227-A requirements. Potential enhancements noted by the ISG further evaluation would be addressed by an AMP enhancement. Proposed Change: Delete further evaluation 3.1.2.2.9.A item 7. Item to be addressed by AMP element 3.	The staff agrees with the comment that VT-3 inspection requirements should be addressed as part of GALL Report AMP XI.M16A. As a result of the staff's resolution of Source ID I-4, final LR-ISG-2011-04 does not incorporate SRP-LR Sections 3.1.2.2.9.A.7, 3.1.2.2.9.C.1, and 3.1.2.2.9.C.4 related to VT-3 inspections. In addition, the staff's position on the use of VT-3 for the detection of cracking will continue to be documented in the "Detection of Aging Effects" program element in GALL Report AMP XI.M16A.
52	II-37	Page A-21 – Section 3.1.2.2.9.B.2 For Westinghouse Hold Down Springs, further evaluation is redundant and inconsistent with standard GALL AMR and AMP formatting and presentation. Definition of physical measurement techniques for Westinghouse hold down springs should be addressed as part of AMP element 3. Acceptance criteria for the hold down spring inspections would be addressed by AMP element 6. Proposed Change: Delete further evaluation 3.1.2.2.9.B item 2. Item to be addressed by AMP elements 3 and 6.	The staff agrees with the comment that physical measurement techniques and the inspection acceptance criteria for Westinghouse hold down springs are to be defined in an AMP. The staff's SE, Revision 1, for MRP-227 documents the basis for limitations and conditions placed on the use of MRP-227 as well as licensee/applicant action items that shall be addressed by applicants/licensees who choose to implement the NRC-approved version of MRP-227. Specifically, A/LAI No. 5 of MRP-227-A addresses physical measurements of Westinghouse hold down springs. As a result of the staff's resolution of Source ID I-1, areas resolved in the

#	Source ID	Summary of Comment	Response
			staff's SE, Revision 1, for MRP-227 and A/LAIs are not addressed in the Further Evaluation sections of the SRP-LR in final LR-ISG-2011-04. In addition, final LR-ISG-2011-04 recommends that license renewal applicants for PWRs provide their responses to the A/LAIs for MRP-227-A in Appendix C of the LRA. As a result of these revisions, final LR-ISG-2011-04 does not incorporate SRP-LR Section 3.1.2.2.9.B.2.
53	II-38	Page A-59 – IV.B2.RP-297 For CASS CRGT Lower Flanges, the ISG revision to the stainless steel definition in GALL Section IX.C requires that CASS be specifically designated in an AMR line item when thermal and neutron embrittlement susceptibility are identified. MRP-227-A Table 3-3 identifies the material of construction for CRGT lower flanges as CF-8 and thermal and neutron embrittlement identified as considerations for primary component classification. Proposed Change: Identify CASS as an additional material in GALL IB.B2.RP-297	The staff agrees with the comment to add cast austenitic stainless steel (CASS) as a material in GALL AMR Item IV.B2.RP-297. In final LR-ISG-2011-04 the "Material" column of GALL AMR Item IV.B2.RP-297 states "stainless steel, including CASS" and the "Aging Effect/Mechanism" column states "Loss of preload due to neutron irradiation embrittlement, and for CASS due to thermal aging embrittlement."
54	II-39	Page A-73 – IV.B2.RP-268 It appears that the primary purpose for the Inaccessible Locations AMR line item is to provide a further evaluation of inaccessible locations in partially accessible components susceptible to cracking due to SCC and IASCC using further evaluation note 3 (SRP-LR Section 3.1.2.2.9A Part A). This further evaluation is redundant to the note 3 further evaluation required by other AMR lines. Proposed Change: Delete IV.B2.RP-268	The staff agrees with the comment to delete IV.B2.RP-268. As a result of the staff's resolution of Source ID I-7 and ID I-8, the format of GALL Tables IV.B2, IV.B3, and IV.B4 in final LR-ISG-2011-04 is consistent with AMR items in the GALL Report for non-RVI components. In addition, the footnotes in the "Further Evaluation" column of these tables are not incorporated into final LR-ISG-2011-04. GALL AMR Items IV.B2.RP-268, IV.B3.RP-309 and IV.B4.RP-238 for Westinghouse, Combustion Engineering and Babcock and Wilcox designed plants, respectively, are not incorporated in final LR-ISG-2011-04.
55	II-40	Page A-73 – IV.B2.RP-269 It appears that the primary purpose for the Inaccessible Locations AMR line item is to provide a further evaluation of inaccessible locations in partially accessible components susceptible to Loss of fracture toughness due to neutron and irradiation embrittlement using further evaluation note 3 (SRP-LR Section 3.1.2.2.9A Part A). This further evaluation is redundant to the note 3 further evaluation required by other AMR lines Proposed Change: Delete IV.B2.RP-269	The staff agrees with the comment to delete IV.B2.RP-269. As a result of the staff's resolution of Source ID I-7 and ID I-8, the format of GALL Tables IV.B2, IV.B3, and IV.B4 in final LR-ISG-2011-04 is consistent with AMR items in the GALL Report for non-RVI components. In addition, the footnotes in the "Further Evaluation" column of these tables are not incorporated into final LR-ISG-2011-04. As a result, GALL AMR Items IV.B2.RP-269, IV.B3.RP-311 and IV.B4.RP-239 for Westinghouse, Combustion Engineering and Babcock and Wilcox designed plants, respectively, are not incorporated into final LR-ISG-2011-04.

#	Source ID	Summary of Comment	Response
56	II-41	Page A-74 – IV.B2.RP-265 No additional measures (Cracking due to SCC and IASCC) in Section 3.3.1 of MRP-227-A defines the no additional measures category as: those PWR internals for which the effects of all eight aging mechanisms are below the screening criteria. Additional components were placed in the No Additional Measures group as a result of FMECA and the functionality assessment. No further action is required by the MRP-227-A for managing the aging of the No Additional Measures components. Simply put, there are no aging effects requiring aging management. Proposed Change: Change the aging effect column and AMP column for IV.B2.RP-265 to be consistent with other GALL AMR "none-none" line items and move the lines to GALL Section IV.E, Common Miscellaneous Material Environment Combinations.	The staff does not agree with the comment to change GALL AMR Item IV.B2.RP-265 to be consistent with other GALL AMR "none-none" line items and the statement that there are no aging effects requirement management. The "No Additional Measures" category of components in MRP-227-A does not equate to such components not having an aging effect requiring management; it only indicates that MRP-227-A does not include guidance to manage aging for components categorized as "No Additional Measures." Thus, the staff agrees with the commenter's following statement that "[n]o further action is required by MRP-227-A for managing the aging of the No Additional Measures components." The IPA is independent of MRP-227-A and may identify applicable aging effects to manage, which may be broader than the aging effects identified in MRP-227-A for RVIs. Thus, the "No Additional Measures" category of components in MRP-227-A does not alleviate the requirements in 10 CFR 54.21(a)(3). In any event, the staff acknowledges that GALL AMR Items IV.B2.RP-265, IV.B2.RP-267, IV.B3.RP-306, IV.B3.RP-307, IV.B4.RP-236 and IV.B4.RP-237 caused confusion; thus, final LR-ISG-2011-04 does not incorporate GALL AMR Items IV.B3.RP-307, IV.B4.RP-236 and IV.B4.RP-237. In addition, GALL AMR Items IV.B2.RP-265, IV.B2.RP-267 and IV.B3.RP-306 in final LR-ISG-2011-04 clarify that there is no additional aging management for reactor internal "No Additional Measures" components unless required by ASME Section XI, Examination Category B-N-3 or relevant operating experience exists.
57	II-42	Page A-74 – IV.B2.RP-267 No additional measures (Loss of fracture toughness due to neutron and irradiation embrittlement) in Section 3.3.1 of MRP-227-A defines the no additional measures category as: those PWR internals for which the effects of all eight aging mechanisms are below the screening criteria. Additional components were placed in the No Additional Measures group as a result of FMECA and the functionality assessment. No further action is required by the MRP-227-A for managing the aging of the No Additional Measures components. Simply put, there are no aging effects requiring aging management. Proposed Change: Change the aging effect column and AMP column for IV.B2.RP-267 to be consistent with other GALL AMR "none-none" line items and move the lines to GALL Section IV.E,	The staff does not agree with the comment to change GALL AMR Item IV.B2.RP-267 to be consistent with other GALL AMR "none-none" line items and that there are no aging effects requirement management. As a result of the staff's resolution for Source ID II-41, final LR-ISG-2011-04 does not incorporate GALL AMR Items IV.B3.RP-307, IV.B4.RP-236 and IV.B4.RP-237. In addition, GALL AMR Items IV.B2.RP-265, IV.B2.RP-267 and IV.B3.RP-306 in final LR-ISG-2011-04 clarify that there is no additional aging management for reactor internal "No Additional Measures" components unless required by ASME Section XI, Examination Category B-N-3 or relevant operating experience exists.

#	Source ID	Summary of Comment	Response
		Common Miscellaneous Material Environment Combinations.	
58	II-43	Page A-6 Clarification is needed relative to the relationship between the SRP-LR and the GALL documents.	The staff noted draft LR-ISG-2011-04 caused confusion between the relationship of the SRP-LR and the GALL Report for PWR RVI components. As a result, final LR-ISG-2011-04 does not reference the SRP-LR in GALL Report AMP XI.M16A in order to be consistent with the format of other AMPs in the GALL Report.
59	II-44	Page A-11 Wording awkward Proposed Change: Delete "that" at the beginning of line 8.	The staff agrees with the comment to delete the word "that" from the "Confirmation Process" program element of GALL Report AMP XI.M16A in draft LR-ISG-2011-04. The staff revised this program element to state, in part, "… for confirming the quality of inspections, flaw evaluations, and corrective actions performed under this program."
60	II-45	Page A-17 There is a concern that "monitoring and trending" program elements and "corrective action" program elements are buried in the Acceptance Criteria section.	The staff agrees with the comment that there is a concern the "monitoring and trending" and "corrective actions" program elements are buried in the Acceptance Criteria section. As a result of the staff's resolution of Source ID I-1 and II-8, the staff revised LR-ISG-2011-04 so that areas resolved in the staff's SE, Revision 1, for MRP-227 and A/LAIs are not addressed in the Further Evaluation sections of the SRP-LR. As a result of these revisions, final LR-ISG-2011-04 does not incorporate SRP-LR Section 3.1.2.2.9.A.4.
61	II-46	Page A-20 and A-21 The statement "To satisfy the requirements of ASME Code Section III…." is confusing if not all plants are committed to Subsection NG. Proposed Change: The statement should be modified to include a qualifying statement like "if the plant is committed to Subsection NG."	The staff does not agree with the comment to alter the referenced statement, as it comes from the staff's SE, Revision 1, on MRP-227. The topic of environmentally-assisted fatigue for PWR RVIs is addressed in A/LAI No. 8, Item 5 of MRP-227-A. Section 3.0 of the staff's SE, Revision 1, on MRP-227 documents the basis for limitations and conditions being placed on the use of MRP-227 as well as licensee/applicant action items that shall be addressed by applicants/licensees who choose to implement the NRC-approved version of MRP-227. Revisions to the conditions and limitations, applicant/licensee plant-specific action items, and conclusions of the staff's SE, Revision 1, for MRP-227 are not within the scope of LR-ISG-2011-04. However, as a result of the staff's resolution of Source ID I-1, areas resolved in the staff's SE, Revision 1, for MRP-227 and A/LAIs are not addressed in the Further Evaluation sections of the SRP-LR in final LR-ISG-2011-04. In addition, final LR-ISG-2011-04 recommends that license renewal applicants for PWRs provide their responses to the A/LAIs for MRP-227-A in Appendix C of the LRA. As a result of these revisions, final LR-ISG-2011-04 does not incorporate SRP-LR Section 3.1.2.2.9.A.9.

#	Source ID	Summary of Comment	Response
62	II-47	Page A-25 – Section 3.1.2.2.9.D.1 The intended meaning of the word "appropriately" in D.1, second paragraph. Is not clear. Proposed Change: Clarify meaning	The staff agrees that the referenced sentence in SRP-LR Section 3.1.2.2.9.D.1 is not clear. However, as a result of the staff's resolution of Source ID I-1, areas resolved in the staff's SE, Revision 1, for MRP-227 and A/LAIs are not addressed in the Further Evaluation sections of the SRP-LR in final LR-ISG-2011-04. As a result of these revisions, final LR-ISG-2011-04 does not incorporate SRP-LR Section 3.1.2.2.9.D.1.
63	II-48	Page A-142 Spell out variable name in 1.0 of "Further Evaluation Recommendations"	The staff noted that as a result of the resolution to Source ID I-7, final LR-ISG-2011-04 does not incorporate the referenced variable names and further evaluation footnotes in GALL Tables IV.B2, IV.B3, and IV.B4.
64	II-49	Page A-5 Each of the following documents provide information for submittal of an AMP and inspection plan: • GALL Revision 2 (page XI.M16A-2) and LR-ISG-2011-04 (draft, page A-5) • RIS 2011-07 (page 6) • Safety Evaluation Revision 1 for MRP-227 (page 34) • Section 3.5.1 of the Safety Evaluation (page 25) It is unclear what actually goes into the LRA and the format. The above verbiage implies that the AMP and inspection plan are separate documents that are submitted with the application but are reviewed and approved by the NRC as unique documents. A quick search of the GALL indicates that PWR Vessel Internals is the only program that requires the AMP and an inspection plan to be submitted for NRC review and approval. Proposed Change: Commenter provided revisions to Section 3.5.1 of Safety Evaluation Revision 1 for MRP-227.	The staff disagrees with the comment because revisions to the conditions and limitations, applicant/licensee plant-specific action items and conclusions of the staff's SE, Revision 1, for MRP-227 are not within the scope of LR-ISG-2011-04. As a result of the staff's resolution of Source ID I-1, areas resolved in the staff's SE, Revision 1, for MRP-227 and A/LAIs are not addressed in the Further Evaluation sections of the SRP-LR in final LR-ISG-2011-04. In addition, final LR-ISG-2011-04 recommends that license renewal applicants for PWRs provide their responses to the A/LAIs for MRP-227-A in Appendix C of the LRA. In addition, see the staff's resolution of Source ID I-3, in which the staff discusses the staff's position/guidance regarding inspection plans that is documented in RIS 2011-07 dated July 21, 2011. The staff expects that the details of an "inspection plan" for Category D plants (defined in RIS 2011-07) will be incorporated into the LRA submittal as part of the 10-element AMP and AMR line items. Thus, consistent with RIS 2011-07, the staff does not expect Category D plants to provide a separate document that contains an "inspection plan" in response to A/LAI No. 8.
65	II-50	Page A-6 GALL Rev 2 (page XI M16A-3) states: The responses to the LR A/LAIs on MRP-227 are provided in Appendix C of the LRA. LR-ISG-2011-04 (page A-6) deleted this requirement, however LR-ISG-2011-04 (page A-14, 15) states to provide responses to the A/LAIs in Appendix C of the LRA, and to address SRP-LR further evaluation "acceptance criteria" that are based on these A/LAIs. It is	The staff agrees with the comment that it is unclear where the A/LAIs should be addressed. As a result of the staff's resolution of Source ID I-1, areas resolved in the staff's SE, Revision 1, for MRP-227 and A/LAIs are not addressed in the Further Evaluation sections of the SRP-LR in final LR-ISG-2011-04. In addition, final LR-ISG-2011-04 recommends that license renewal applicants for PWRs provide their responses to the A/LAIs for MRP-227-A in Appendix C of the LRA. As a result of these revisions, final LR-ISG-2011-04 does not incorporate SRP-LR Section 3.1.2.2.9.A.1 and also does not incorporate a discussion of A/LAIs in GALL AMP XI.M16A.

#	Source ID	Summary of Comment	Response
		unclear where the licensee action items should be addressed. Wording implies that the applicant/licensee action items should be addressed in Appendix C and in the associated further evaluation section. Proposed Change: The Commenter provided revisions to LR-ISG-2011-04 (page A-6).	
66	II-51	XI.M16A, PWR Vessel Internals elements 1. Scope of Program, 5. Monitoring and Trending, and 6. Acceptance Criteria refer to the latest NRC approved version of WCAP-17096-NP and the associated applicant/licensee action items. It is our understanding that WCAP-17096-NP has been submitted for approval however it has not been approved at this time. A program cannot be developed based on an unapproved document or unknown A/LAIs. Proposed Change: Remove any reference to WCAP-17096-NP or delay issuance of LR-ISG-2011-04 until WCAP-17096-NP is approved by the NRC.	The staff agrees with the comment to delete any reference to WCAP-17096-NP since the report has been submitted for review but not approved by staff. Final LR-ISG-2011-04 does not reference WCAP-17096-NP.
67	II-52	Many of the A/LAIs specified in the Acceptance Criteria section of LR-ISG-2011-04 request that the applicant make enhancements or augmented enhancements to various program elements as a result of the responses to the "further evaluations." It would be simpler if the NRC specified an acceptable method of addressing an issue in the XI.M16A program elements and then if the licensee/applicant wanted to do something different take an exception rather than requiring each licensee/applicant to develop a unique set of enhancements for their program. Proposed Change: Revise A/LAIs to clearly state that additional justification/information is only required to be included in the "further evaluation" responses if the applicant/licensee is deviating from the requirements of MRP-227-A.	The staff disagrees with the comment to revise A/LAIs in LR-ISG-2011-04, as it is a direct reference to the staff's SE, Revision 1, for MRP-227. See the staff's resolution to Source ID II-49, in which the staff explains that revisions to the conditions and limitations, applicant/licensee plant-specific action items and conclusions of the staff's SE, Revision 1, for MRP-227 are not within the scope of LR-ISG-2011-04. No revisions were made as a result of this comment.
68	II-53	A simplified method of addressing reactor internals in the GALL tables B.2, B.3, and B.4 would be to have line items based on component classification (Primary, Expansion, Existing, and No Additional Measures) as defined in MRP-227-A rather than individual component types (Alignment and Interfacing components: internals hold down spring, Alignment and interfacing components: upper core plate alignment pins, etc). Making this change would allow multiple line items to apply to several component types and	The staff agrees with the comment, in part, that GALL AMR Tables IV.B2, IV.B3 and IV.B4 can be simplified. However, the staff does not agree with the commenter's proposed change. As explained in the staff's resolution of Source ID I-6, the AMR line items in the GALL Report and MRP-227-A do not solely serve as the basis for determining components or aging effects that require management or establish the AMR line items to be included in an LRA. The IPA required

#	Source ID	Summary of Comment	Response
		reduce the number of Table 2 line items simplifying this section. Proposed Change: Revise NUREG-1801 tables B.2, B.3, and B.4 to have line items associated with component classification (Primary, Expansion, Existing Program, and No Additional Measures) and refer to MRP-227-A for the specific components in the four classification groups.	by 10 CFR 54.21(a) is independent of the AMR line items provided in MRP-227-A and the GALL Report. It is not necessary for the staff to correlate the number and contents of AMR items in GALL Tables IV.B2, IV.B3, and IV.B4 exactly to the number and contents of inspection items in MRP-227-A. In any event, as a result of the staff's resolution of Source ID I-1, GALL AMR Tables IV.B2, IV.B3 and IV.B4 were revised. See resolution of Source ID I-1 for a summary of the revisions.
69	II-54	Several of the applicant/licensee action items (A/LAI) identified in the Safety Evaluation for MRP-227 (pages 32 – 34) required plant-specified evaluations or analyses to be submitted as part of the application. A/LAI Number 5 requires the applicant/licensee include its proposed acceptance criteria and an explanation of how the proposed acceptance criteria are consistent with the plants' licensing basis and the need to maintain the functionality of the component being inspected under all licensing basis conditions of operation during the period of extended operation as part of their submittal to apply the approved version of MRP-227. A/LAI Number 7 requires a plant-specific analysis to be performed on Westinghouse lower support column bodies made of CASS be included as part of their submittal to apply the approved version of MRP-227. Proposed Change: Revise A/LAI Numbers 5 and 7 to allow for the applicants/licensees to commit performing an analysis prior to the period of extended operation. This would allow applicants/licensees that are submitting in the near future (2013 timeframe) to perform the analyses on normal schedule rather than an expedited schedule.	The staff disagrees with the comment to revise A/LAIs in LR-ISG-2011-04, as it is a direct reference to the staff's SE on MRP-227-A. See the staff's resolution to Source ID II-49, in which the staff explains that revisions to the conditions and limitations, applicant/licensee plant-specific action items and conclusions of the staff's SE, Revision 1, for MRP-227 are not within the scope of LR-ISG-2011-04. No revisions were made as a result of this comment.
70	II-55	Page A-17 In the last paragraph it states: For those component inspections that do not achieve the inspection coverage criteria stated in the NRC SE (Revision 1) on MRP-227, the applicant is requested to identify a deviation from the MRP-defined inspection criteria and describe the process and type of evaluation that will be implemented to evaluate the impact of the aging effects on the integrity of those components in the population that were inaccessible to the inspection technique, and to identify this process as an applicable enhancement of the "monitoring and trending" program element of its RVI Program.	The staff agrees with the comment that the referenced statement in SRP-LR Section 3.1.2.2.9.A.4 of draft LR-ISG-2011-04 is more appropriately addressed in the AMP. As a result of the staff's resolution of Source ID I-1, areas resolved in the staff's SE, Revision 1, for MRP-227 and A/LAIs are not addressed in the Further Evaluation sections of the SRP-LR in final LR-ISG-2011-04. As a result of these revisions, final LR-ISG-2011-04 does not incorporate SRP-LR Section 3.1.2.2.9.A.4.

#	Source ID	Summary of Comment	Response
		The SRP is specifying actions for applicants to perform as part of an aging management program which is more appropriately addressed within program requirements. Proposed Change: These actions should be included as program elements, not in the further evaluation sections of the SRP.	
71	II-56	Page A-19 and A-20 – Section 3.1.2.2.9.A.9 In the third paragraph the further evaluation states: "To satisfy the requirements of the ASME Code, Section III, Subsection NG-2160 and NG-3121, the existing fatigue CUF analysis shall include the effects of the reactor coolant water environment." The December 26, 1999, Generic Safety Issue (GSI) 190 close-out memorandum from Ashok C. Thadani, Director of the Office of Regulatory Research, to William D. Travers, Executive Director for Operations, provides the basis for consideration of the effects of the reactor coolant water environment. It should be noted that GSI-190 concerns are limited to pipe leakage, which is not applicable to RVI components since they do not form a portion of the reactor coolant pressure boundary and are therefore not subject to leakage. Proposed Change: Delete the referenced sentence in the third paragraph of Further Evaluation A.9.	The staff does not agree with the comment, in particular the inference that, the effects of the reactor coolant water environment on metal fatigue are not applicable to RVI components since they do not form a portion of the reactor coolant pressure boundary. See the staff's resolution to Source ID II-46, in which the staff explains that environmentally-assisted fatigue for PWR RVIs is addressed specifically in A/LAI No. 8, Item 5 of the staff's SE, Revision 1, on MRP-227. Final LR-ISG-2011-04 recommends that license renewal applicants for PWRs provide their responses to the A/LAIs for MRP-227-A in Appendix C of the LRA. As a result of these revisions, final LR-ISG-2011-04 does not incorporate SRP-LR Section 3.1.2.2.9.A.9
72	II	The comments submitted by EPRI-MRP, the PWROG-MSC, and NEI are extensive and involve complex issues. EPRI and PWROF, along with NEI, respectfully request a follow-up meeting with the NRC staff to discuss resolution of the comments and, if appropriate, an additional comment period.	The NRC staff acknowledges the complexity of the issues captured in MRP-227-A. However, LR-ISG-2011-04 is not intended to supplement, modify or further resolve the issues raised in MRP-227-A, but rather to reference MRP-227-A and the associated staff's SE, Revision 1, for MRP-227, in the usable format of a generic aging management program, as described in the GALL Report. To the extent that comments provided suggestions to clarify or simplify the format of LR-ISG-2011-04 for ease of use, the staff was able to incorporate those changes into the final document. However, to the extent that comments proposed changes to the actual content of the staff's SE, Revision 1, for MRP-227, the staff did not incorporate those comments, as it is beyond the scope and intent of LR-ISG-2011-04. The staff also does not perceive further benefits from an additional public meeting and comment period to resolve the latter set of comments, as it is beyond the scope of LR-ISG-2011-04.