Appendix A

Staff Proposed Revisions to NUREG-1800, Revision 2, “Standard Review Plan for License Renewal Applications for Nuclear Power Plants” and NUREG-1801, Revision 2, “Generic Aging Lessons Learned”

This appendix provides the staff's proposed changes to the recommended guidance for pressurized-water reactor (PWR) reactor vessel internal (RVI) components in NUREG-1800, Revision 2, "Standard Review Plan for License Renewal Applications for Nuclear Power Plants" (SRP-LR, Revision 2) and NUREG-1801, Revision 2, "Generic Aging Lessons Learned" (GALL, Revision 2) Report. For the purposes of this appendix, reference to MRP-227 is the reference to Electric Power Research Institute (EPRI) Report No. 1016596, "Materials Reliability Program: Pressurized Water Reactor Internals Inspection and Evaluation Guidelines (MRP-227-Rev. 0)," December 2008, as supplemented by relevant information provided by the MRP in its responses to applicable NRC requests for additional information (RAIs) on the contents of the report. Reference to MRP-227-A is the reference to EPRI Report No. 1022863, "Materials Reliability Program: Pressurized Water Reactor Internals Inspection and Evaluation Guidelines (MRP-227-A)," December 2011, as approved in Revision 1 of the staff's safety evaluation (i.e., the NRC SE (Rev. 1) on MRP-227 [ADAMS ML11308A770]) on the report.

The staff's proposed changes to the GALL Report, Revision 2 and the SRP-LR, Revision 2, are given in the following sections of this appendix:

- Section 1 – proposed changes to GALL aging management program (AMP) XI.M16A, "PWR Vessel Internals"

- Section 2 – proposed changes to the NRC's further evaluation "acceptance criteria" and "review procedure" recommendations for PWR RVI components in Sections 3.1.2.2 and 3.1.3.2 of the SRP-LR, Revision 2, and to the Final Safety Analysis Report (FSAR)/Updated FSAR (UFSAR) Supplement example for the PWR Vessel Internals Program in Table 3.0-1 of the SRP-LR, Revision 2

- Section 3 – proposed changes to the commodity group-based AMR items for PWR RVI components in Table 3.1-1 of the SRP-LR, Revision 2

- Section 4 – proposed changes to the component-specific AMR items for Westinghouse-designed PWR RVI components in Table IV.B2 of the GALL Report, Revision 2

- Section 5 – proposed changes to the component-specific AMR items for Combustion Engineering-designed (CE-designed) PWR RVI components in Table IV.B3 of the GALL Report, Revision 2

- Section 6 – proposed changes to the component-specific AMR items for Babcock and Wilcox-designed (B&W-designed) PWR RVI components in Table IV.B4 of the GALL Report, Revision 2

- Section 7 – proposed changes to the recommended definition for "stainless steel" materials in the GALL Table IX.C, "Selected Definitions & Use of Terms for Describing and Standardizing – MATERIALS"

With the exception of the changes proposed to the SRP-LR-defined "further evaluation" recommendations in Section 2, all proposed changes to applicable sections and tables of the GALL Report, Revision 2, and SRP-LR, Revision 2, are identified by the following formatted text: (a) proposed deletions of existing text are identified by "black strikeout formatted" text, and (b) proposed additions to existing text are identified by "black double underlined formatted" text. However, the following additional clarifications are necessary.

The staff drafted the current version of GALL AMP XI.M16A in the GALL Report, Revision 2, in a manner that could be adopted by a PWR applicant applying for license renewal of its facility. AMP text that could be copied by the applicant for incorporation into its PWR Vessel Internals AMP was identified in “black” colored text in the AMP. Plant-specific text that would need to be written by the applicant for the program elements of its PWR Vessel Internals Program was identified in “black italicized” text. However, with the exception of the first sentence in the “scope of program” element in GALL AMP XI.M16A, the updated version of GALL AMP XI.M16A in this appendix no longer includes “italicized text” that would prompt the applicant into providing plant-specific administrative information entries into the program elements for its AMP.

For the update of the SRP-LR “further evaluation” sections, the staff recognized that the NRC SE (Rev. 1) on MRP-227 occurred after the issuance of the GALL Report, Revision 2, and included applicable applicant/licensee action items (A/LAIs) and topical report condition items (TRCIs). The staff’s intent is to write an updated set of further evaluation “acceptance criteria” recommendations that would aid a PWR license renewal applicant in identifying the plant-specific aspects of its AMP. The updated recommendations are based on the relevant A/LAIs, TRCIs or specific program element criteria in GALL AMP XI.M16A.

The staff decided to consolidate the further evaluation “acceptance criteria” recommendations for PWR RVI components into one section of the SRP-LR, Revision 2 (SRP-LR Section 3.1.2.2.9), with applicable subsections. Some of these subsections incorporate and supersede the staff’s further evaluation “acceptance criteria” recommendations for PWR RVI components that were previously given in SRP-LR, Revision 2, Sections 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. However, in consideration of the A/LAIs and TRCIs that were identified in the NRC SE (Rev. 1) on MRP-227, this license renewal interim staff guidance (LR-ISG) identifies a number of new further evaluation recommendations for PWR RVI components that were previously unaccounted for in the SRP-LR, Revision 2.

Similarly, the staff decided to consolidate the further evaluation “review procedure” recommendations for PWR RVI components into a new SRP-LR, Revision 2, further evaluation “review procedure” section that supersedes the staff’s previous “review procedure” recommendations for PWR RVI components in SRP-LR, Revision 2, Sections 3.1.3.2.9, 3.1.3.2.10, 3.1.3.2.12, 3.1.3.2.13, and 3.1.3.2.14. The new SRP-LR section is written as an update of SRP-LR, Revision 2, Section 3.1.3.2.9.

Appendix A – Section 1: Staff Proposed Revision to AMP XI.M16A, "PWR Vessel Internals," in the Generic Aging Lesson Learned Report, Revision 2 (GALL Report, Revision 2)

XI.M16A PWR VESSEL INTERNALS

Program Description

This program relies on implementation of the Electric Power Research Institute (EPRI) Materials Reliability Program (MRP) Report No. ~~1016596~~1022863 (MRP-227-A) and EPRI Report No. 1016609 (MRP-228) to manage the aging effects on the pressurized-water reactor (PWR) reactor vessel internal (RVI) components. The MRP recommended activities in MRP-227-A and any additional plant-specific activities that need to be defined for this program are implemented in accordance with the guidelines of Nuclear Energy Institute (NEI) 03-08. The staff approved the MRP's augmented inspection and evaluation (I&E) criteria for PWR RVI components in a safety evaluation (i.e., the NRC SE (Rev. 1) on MRP-227) 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/cyclical 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 dimension due to void swelling; 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 NRC Branch Technical Position RSLB-1, with periodic examinations and other inspections of highly-affected internals locations. These examinations provide reasonable assurance that the effects of age-related degradation mechanisms will be managed during the period of extended operation. The program includes expanding periodic examinations and other inspections, if the extent of the degradation effects exceeds the expected levels.

The MRP-227-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 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 internals component locations for each of the three plant designs that are expected to show the leading indications of the degradation effects, with another set of expansion internals component locations that are specified to expand the sample should the indications be more severe than anticipated.
The degradation effects in a third set of internals locations are deemed to be adequately managed by Existing Programs, such as American Society of Mechanical Engineers

(ASME) Code, Section XI,[~~11~~1] Examination Category B-N-3, examinations of core support structures. A fourth set of internals locations are deemed to require no additional measures. As a result, the program typically identifies 5 to 15 percent of the RVI locations as Primary Component locations for inspections, with another 7 to 10 percent of the RVI locations to be inspected as Expansion Components, as warranted by the evaluation of the inspection results. Another 5 to 15 percent of the internals locations are covered by Existing Programs, with the remainder requiring no additional measures. 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-A 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-A guidance for visual examinations goes beyond the ASME Code, Section XI visual examination criteria because additional guidance is incorporated into MRP-227-A 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 the baffle-former bolts in Babock and Wilcox (B&W)-designed units and Westinghouse-designed units, has been well demonstrated by operating experience.

In addition, the program's adoption of the MRP-227-A 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-A. 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 in the United States of America (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 the 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.

1~~1~~ Refer to the GALL Report, Chapter I, for applicability of various editions of the ASME Code, Section XI.

Evaluation and Technical Basis

1. ***Scope of Program:*** The scope of the program includes all RVI components at the [*as an administrative action item for the aging management program (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, Combustion Engineering (CE), or B&W, as applicable*] nuclear steam supply system (NSSS) design. The scope of the program applies the methodology and guidance in MRP-227-A (i.e., the most recently NRC-endorsed version of MRP-227), which provides augmented inspection and flaw evaluation methodology for assuring the functional integrity of safety-related internals in commercial operating U.S. PWR nuclear power plants designed by B&W, CE, and Westinghouse. The scope of components considered for inspection under MRP-227-A guidance includes 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). 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 responses ~~bases~~ to applicable ~~license renewal applicant action items~~ applicant/licensee action items (~~LRAAIs~~A/LAIs) on the MRP-227-A methodology (as identified in Rev. 1 of the NRC SE on MRP-227), and to applicable subsections in Section 3.1.2.2 of NUREG-1800, Revision 2 (i.e., SRP-LR "further evaluation" criteria sections), and any additional programs, actions, or activities that are discussed in these A/LAI or "further evaluation" responses and are credited for aging management of the applicant's RVI components.[2] ~~The LRAAIs are identified in the staff's safety evaluation onMRP-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.~~ Additional criteria are provided in the latest NRC-endorsed version of Westinghouse Commercial Atomic Power (WCAP) Report No. WCAP-17096-NP, "Reactor Internals Acceptance Criteria Methodology and Data Requirements". If WCAP-17096-NP is used as a basis for superseding applicable criteria in MRP-227-A, the application of the report is subject to

[2] Consistent with the NRC recommendations in the applicable SRP-LR "further evaluation" sections, those A/LAI response bases or SRP-LR "further evaluation" response bases that result in the need for augmentation of the AMP beyond the I&E criteria recommended in the MRP-227-A report, are requested to be identified as plant-specific enhancements of the AMP, along with the applicable program element criteria that they impact. Justifications for the enhancements are to be provided in the applicable responses to the A/LAI items or SRP-LR "further evaluation" items.

the conditions and A/LAIs established in the staff's SE endorsing the use of the WCAP report.

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. 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/cyclical 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 dimension due to void swelling and irradiation growth, distortion, or deflection; and (e) loss of preload caused by thermal and irradiation-enhanced stress relaxation or creep. For the management of cracking, the program monitors for evidence of surface breaking linear discontinuities if a visual inspection technique is used as the non-destruction examination (NDE) method, or for relevant flaw presentation signals if a volumetric UT method is used as the NDE method. For the management of loss of material, the program monitors for gross or abnormal 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~~;~~. ~~i~~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 cracking is detected in the components and is extensive enough to warrant a supplemental flaw growth or flaw tolerance evaluation ~~under the MRP-227-A guidance or ASME Code, Section XI requirements~~. The program uses physical measurements to monitor for any dimensional changes due to void swelling, irradiation growth, distortion, or deflection.

Specifically, the program implements the parameters monitored/inspected criteria consistent with the applicable condition monitoring tables in MRP-227-A, as applicable to the nuclear steam system supply vendor for the plant's RVI components, or as enhanced based on the applicant's response bases to applicable A/LAIs in Rev. 1 of the NRC SE on MRP-227, or to applicable NRC further evaluation recommendations in the SRP-LR (Refer to Footnote 2).~~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.~~

4. *Detection of Aging Effects:* ~~The~~ Since the program is consistent with MRP-227-A and MRP-228, the detection of aging effects is ~~covered in two places: (a) the guidance~~introduced in Section 4 of MRP-227-A ~~provides an introductory discussion and justification of the examination methods selected for detecting the aging effects of interest; and (b)~~and standards for the examination methods, procedures, and personnel are provided in a companion document, MRP-228. In all cases, well-established methods were selected. These methods include volumetric UT examination methods for detecting flaws in bolting~~, 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 and the program may apply physical measurement techniques for the detection of loss of preload in fastened or bolted RVI assemblies or changes in dimension as a result of distortion, irradiation-assisted creep or void swelling.

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 if justified in accordance with the NRC's "further evaluation" criteria in SRP-LR Section 3.1.2.2.9.A.7, which provides the NRC's further evaluation "acceptance criteria" recommendations on this matter.~~only when the flaw tolerance of the component or affected assembly, as evaluated for reduced fracture toughness properties, is known and has been shown to be tolerant of easily detected large flaws, even under reduced fracture toughness conditions.~~

In addition, VT-3 examinations are used to monitor/inspect for loss of material induced by wear and for general aging conditions, such as gross distortion caused by void swelling and irradiation growth or by gross effects of loss of preload caused by thermal and irradiation-enhanced stress relaxation and creep.

In addition, the program adopts the recommended guidance in MRP-227-A for defining the Expansion criteria that need to be applied to inspections of Primary Components and Existing Requirement Components and for expanding the examinations to include additional Expansion Components. ~~As a result, inspections performed on the~~ RVI component inspections are performed consistent with the inspection frequency and sampling bases for Primary Components, Existing Requirement Components, and Expansion 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-227or "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), physical measurements are used as supplemental techniques to manage for the gross effects of wear, loss of preload due to stress relaxation, or for changes in dimension due to void swelling, deflection or distortion. The physical measurements methods applied in accordance with this program include~~ [~~*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*~~]~~.~~

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, including~~e~~ recommendations for flaw depth sizing and for crack growth determinations as well for performing applicable limit load, linear elastic and elastic-plastic fracture analyses of relevant flaw indications, are given in MRP-227-A, as supplemented by information in the latest NRC-endorsed version of the WCAP-17096-NP report. The examinations and re-examinations required by the MRP-227-A guidance, together with the flaw evaluation criteria in the WCAP-17096-NP report and 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~~of the aging effects and mechanisms ~~within the scope of~~ that are managed by the program. ~~The extent of the examinations, beginning with the sample of susceptible PWR internals component locations identified as Primary Component locations, with the potential for inclusion of Expansion Component locations if the effects are greater than anticipated, plus the continuation of the Existing Programs activities, such as the ASME Code, Section XI, Examination Category B-N-3 examinations for core support structures, provides a high degree of confidence in the total program.~~

6. *Acceptance Criteria:* Section 5 and Table 5-1 of MRP-227-A, as supplemented by information in NRC approved versions of WCAP-17096-NP, provides specific examination and flaw evaluation acceptance criteria for the Primary and Expansion Component examination methods, and include those for visual surface and volumetric techniques. For components addressed by examinations referenced to ASME Code, Section XI, the IWB-3500 acceptance criteria apply. For ~~other~~ components covered by Existing Programs, the examination acceptance criteria are described within the Existing Program reference document (e.g., those for Westinghouse-design flux thimble tubes under a Westinghouse applicant's NRC Bulletin 88-09 implementation program).

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 Rev. 1 of the NRC SE on MRP-227.[3]

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

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

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

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

7. *Corrective Actions:* Corrective actions following the detection of unacceptable conditions are fundamentally provided for in each plant's corrective action program. Any detected conditions that do not satisfy the examination acceptance criteria are required to be dispositioned through the plant corrective action program, which may require repair, replacement, or analytical evaluation for continued service until the next inspection. The disposition will ensure that design basis functions of the reactor internals components will continue to be fulfilled for all licensing basis loads and events. ~~Examples of methodologies that can be used to analytically disposition unacceptable conditions are found in the ASME Code, Section XI or in Section 6 of MRP-227. Section 6 of MRP-227 describes the options that are~~

[3] The EPRI MRP's recommendations are applicable to physical measurements of Westinghouse design hold-down springs and to those CE-design nuclear facilities that are designed with welded core shrouds made from two vertical shroud segments. Based on the recommendations, Westinghouse applicants should define, as enhancements to the program, the acceptance criteria that will be applied to physical measurement methods used to manage loss of compressibility (loss of preload) in their hold-down springs, as based on the applicant's response bases to the further evaluation "acceptance criteria" recommendations in SRP-LR Section 3.1.2.2.9.B.2. Similarly, CE applicants whose plants are designed with welded core shrouds containing the specified gap area, should define, as enhancements of the program, the acceptance criteria that will be applied to physical measurement methods used to manage distortion (changes in dimension) in the gap areas of the shrouds, as based on the applicant's response bases to the further evaluation "acceptance criteria" recommendations in SRP-LR Section 3.1.2.2.9.C.2. The MRP did not recommend physical measurement monitoring bases for Babcock and Wilcox (B&W) designed plants or for CE plants with bolted core shroud designs.

~~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-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 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~~, such as the 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 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-A, in conjunction with NEI 03-08 and other guidance documents, reports, or methodologies referenced in this AMP, ~~will~~ provides an acceptable level of quality and an acceptable basis for confirming that the quality of inspections, flaw evaluations, and corrective actions performed under this program.~~other elements of aging management of the PWR internals that are addressed in accordance with the 10 CFR Part 50, Appendix B, or their equivalent (as applicable), confirmation process, and administrative controls.~~

9. ***Administrative Controls:*** The administrative controls for ~~such~~ 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 additionally in accordance with the ~~under~~ existing site 10 CFR 50 Appendix B, Quality Assurance Programs, or their equivalent, as applicable. The evaluation in Section 3.5 of Revision 1 of the SE on the MRP-227 methodology provides the NRC'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 justifying the deviation no later than 45 days after approval by a licensee executive. ~~Such a program is thus expected to be established with a sufficient level of documentation and administrative controls to ensure effective long-term implementation.~~

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

~~The application of the MRP-227 guidance will establish a considerable amount of operating experience over the next few years. Section 7 of MRP-227 describes the reporting requirements for these applications, and the plan for evaluating the accumulated additional operating experience.~~ The review of relevant operating experience (OE) and the assessment of OE for its impacts on the program elements of an applicant's PWR Vessel Internals Program, and on the program's implementing procedures and review and approval processes, are governed by the recommended industry guidelines and criteria in NEI 03-08, and additionally in accordance with the existing site 10 CFR 50 Appendix B Quality Assurance Programs, or their equivalent, as applicable. 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 justifying the deviation no later than 45 days after approval by a licensee executive. As discussed in Appendix B of the GALL Report, the ongoing effectiveness of the program is ensured through the systematic review of both plant-specific and industry operating experience.

Consistent with MRP-227-A, the reporting of PWR RVI inspection and OE results is treated as a "Needed" category item under the applicant's NEI 03-08 implementation process. Based on these criteria, such programs are expected to be established with a sufficient level of documentation and administrative controls to ensure effective long-term implementation.

References

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10 CFR Part 50.55a, *Codes and Standards*, Office of the Federal Register, National Archives and Records Administration, 20~~09~~11.

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Appendix A - Section 2: Staff Proposed Revision to SRP-LR Section 3.1.2.2 Further Evaluation Acceptance Criteria Sections; Section 3.1.3.2 Further Evaluation Review Procedure Sections; and SRP-LR Table 3.0-1, FSAR Supplement for Aging Management of Applicable Systems

Itemized List of Previous SRP-LR Further Evaluation Sections Being Replaced or Superseded

- "Acceptance Criteria" Section 3.1.2.2.9
- "Acceptance Criteria" Section 3.1.2.2.10
- "Acceptance Criteria" Section 3.1.2.2.12
- "Acceptance Criteria" Section 3.1.2.2.13
- "Acceptance Criteria" Section 3.1.2.2.14
- "Review Procedures" Section 3.1.3.2.9
- "Review Procedures" Section 3.1.3.2.10
- "Review Procedures" Section 3.1.3.2.12
- "Review Procedures" Section 3.1.3.2.13
- "Review Procedures" Section 3.1.3.2.14

Updated Draft Further Evaluation "Acceptance Criteria" Sections for PWR RVI Components in SRP-LR Section 3.1.2.2

ACCEPTANCE CRITERIA

3.1.2.2.9 Augmented Inspection Bases for PWR Vessel Internal Components

The AMR item guidance for PWR reactor vessel internal (RVI) components-is based on the augmented inspection and evaluation (I&E) guidance in EPRI Report No. 1022863, "Materials Reliability Program: Pressurized Water Reactor Internals Inspection and Evaluation Guidelines (MRP-227-A)" (NRC ADAMS Accession Nos. ML12017A193 – ML12017A199), as subject to the evaluation, Topical Report Condition Items (TRCIs), Applicant/Licensee Action Items (A/LAIs), and conclusions in Revision 1 of the staff's safety evaluation on the report (i.e., the NRC SE (Rev. 1) on MRP-227), dated December 16, 2011 (NRC ADAMS Accession No. ML11308A770).

The following license renewal further evaluation "acceptance criteria" items apply to the evaluation and aging management basis for PWR RVI components: (a) generic "acceptance criteria" recommendations that are applicable to the RVI components for all PWR nuclear steam supply service (NSSS) designs; (b) "acceptance criteria" recommendations that are applicable to specific AMR items in GALL Table IV.B2 for Westinghouse-design RVI components; (c) "acceptance criteria" recommendations that are applicable to specific AMR items in GALL Table IV.B3 for Combustion Engineering (CE)-design RVI components; and (d) "acceptance criteria" recommendations that are applicable to specific AMR items in GALL Table IV.B4 for Babcock and Wilcox (B&W)-design RVI components.

A. *Further Evaluation "Acceptance Criteria" Recommendations for PWR RVI Components – Generic Items*

1. <u>Response to A/LAIs on the MRP-227 Report</u>:
   The NRC SE (Rev. 1) on MRP-227 includes a number of A/LAIs on the use of the MRP's recommended augmented I&E methodology in the report. Applicant's applying for license renewal of their PWR facilities are requested to provide their specific responses to

the A/LAIs on the MRP-227-A methodology in Appendix C of their LRAs, and to address the information requested in the LRA sections that respond to the specific SRP-LR further evaluation “acceptance criteria” that are based on these A/LAIs.

Section 3.5.1 of the NRC SE (Rev. 1) on MRP-227 (A/LAI No. 8, Item 3) reminds PWR applicants of the need to include applicable FSAR, USAR, or UFSAR Supplement summary descriptions, in accordance with 10 CFR 54.21(d) requirements, for the RVI Program and any associated TLAAs that need to be identified for their RVI components under the requirements of 10 CFR 54.21(c)(1). Consistent with NEI 95-10, Revision 6, the staff recommends that these FSAR, USAR, or UFSAR Supplement summary descriptions be included in Appendix A of the LRA.

2. RVI Program and Inspection Plans for PWR RVI Components: Section 3.5.1 of the NRC SE (Rev. 1) on MRP-227 (A/LAI No. 8, Item 1) requested that applicants include in their LRAs an AMP for the facility that addresses 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). A/LAI No. 8, Item 2 in Section 3.5.1of the NRC SE (Rev. 1) on MRP-227 states that each PWR applicant for renewal should submit an inspection plan for their RVI components that addresses those plant-specific A/LAIs that are applicable to the NSSS-design of their RVI components and to submit the inspection plan for NRC review and approval consistent with the current licensing basis (CLB) for the facility. Consistent with the NEI 03-08 process protocol, the staff also requested that each applicant identify those aspects of its program that deviate from the specific aging management inspections or evaluations recommended in MRP-227-A and provide a justification for any deviations that would impact the inspection criteria for its “Primary Category” and “Expansion Category” components.

To address A/LAI No. 8, Item 1, PWR-design license renewal applicants are requested to include an AMP that corresponds to the program GALL AMP XI.M16A, “PWR Vessel Internals,” as part of the scope of the license renewal AMPs that are included in Appendix B of the LRA. Applicants are requested to assess the need for enhancing the program elements for their RVI Programs, as appropriate, as based on their responses to A/LAIs that are applicable to the design of their RVI components or to the applicable AMR further evaluation sections for the RVI components, as given in this section (i.e., Section 3.1.2.2.9 of the SRP-LR).

To address A/LAI No. 8, Item 2, PWR-design license renewal applicants applying for license renewal of their facilities are requested to include their inspection plans for their RVI components as part of their LRAs.

Consistent with NEI 95-10, Revision 6, for those AMPs or inspection plans that include deviations from the MRP’s proposed I&E methodology, as approved in the NRC SE (Rev. 1) on MRP-227, the applicant is requested to identify each deviation from the MRP recommendations as an exception to the appropriate program element criterion for the affected activity in GALL AMP XI.M16A. The applicant is also requested to justify its basis for concluding that the deviation taken to the applicable MRP protocol, and any proposed alternative to that protocol, will be capable of managing the applicable aging effect and maintaining the intended function(s) of the applicable impacted component or components during the period of extended operation.

3. PWR Vessel Internals Scoping and Inspection Category Review: The guidance in MRP-227-A specifies applicability limitations to base-loaded plant assumptions and the fuel loading management assumptions upon which the MRP's functionality analyses were based. Section 3.5.1 of the NRC SE (Rev. 1) on MRP-227 (A/LAI No. 1) states that each applicant/licensee should refer to the specific assumptions regarding plant design and operating characteristics made in the MRP's Failure Mode, Effects, and Criticality Analysis (FMECA) and functionality analyses for their applicable NSSS reactor design (i.e., Westinghouse, Combustion Engineering [CE], or Babcock and Wilcox [B&W] NSSS designs) and should describe the process that would be used to determine plant-specific differences in the design or operating conditions of their RVI components from those assumed by the MRP and that would result in different component inspection categories from those recommended in the MRP report.

   Section 3.2.5.2 of the NRC SE (Rev. 1) on MRP-227 (A/LAI No. 2) states that each applicant/licensee is responsible for identifying which RVI components are within the scope of license renewal (LR) for its facility. As a result, the staff stated that applicants should review the information in Tables 4-1 and 4-2 in MRP Report No. MRP-189, Revision 1, and Tables 4-4 and 4-5 in MRP Report No. MRP-191 and should identify whether these tables contain all of the RVI components that need to be within the scope of LR for their facilities, as required by 10 CFR 54.4. In A/LAI No. 2, the staff stated that if the tables did not identify all of the RVI components that are within the scope of LR for its facility, the applicant or licensee should identify the missing component(s) and should propose any necessary modifications to the program defined in MRP-227-A when submitting its plant-specific AMP, such that the effects of aging on the missing component(s) would be adequately managed for the period of extended operation in accordance with the aging management requirements of 10 CFR 54.21(a)(3).

   To address these A/LAIs, PWR applicants requesting license renewal of their facilities are requested to perform the confirmatory integrated plant assessment (IPA) review under the process that was described in their A/LAI No. 1 response to determine, as indicated by A/LAI No. 2, whether additional RVI components should be inspected on a plant-specific basis – that is beyond those that would be inspected under the MRP's recommended inspection criteria for "Primary Category," "Expansion Category," and "Existing Program" components in MRP-227-A. 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. The applicant is also requested to identify the changes in the component inspection category criteria in appropriate augmented enhancements of the "scope of program," "detection of aging effects," "monitoring and trending," or "acceptance criteria" program elements for their RVI-Programs, as applicable. This action includes the need to identify any plant-specific RVI component inspections or evaluations that are mandated in the current operating license for the facility or in the plant technical specifications as plant-specific activities for the AMP's "scope of program" element. The applicant is to justify the bases for these enhancements as part of its basis for responding to these further evaluation recommendations.

4. Partially inaccessible components: Further evaluation of inspection coverages and inaccessible areas is recommended in partially accessible MRP-defined "Primary

Category” RVI components, as described in Section 3.3.1 of the NRC SE (Rev. 1) on MRP-227, and for inaccessible areas for MRP-defined “Expansion Category” RVI components in Section 3.3.2 of the NRC SE (Rev. 1) on MRP-227 .

To address the technical basis in Sections 3.3.1 and 3.3.2 of the NRC SE (Rev. 1) on MRP-227 , the applicant is requested to discuss in its response to this further evaluation “acceptance criteria” recommendation how the relevant effects of aging will be evaluated in the inaccessible portions of a non-redundant component, if age-related degradation has been detected in the areas of the component that are accessible to examination, or if the inspection coverage referenced in the NRC SE (Rev. 1) on MRP-227 is not achieved.

Specifically, for those inspections of non-redundant components that achieve the inspection coverage criteria stated in the NRC SE (Rev. 1) on MRP-227, and result in detection of relevant aging effects, the applicant is requested to describe the process and type of evaluation that will be implemented to evaluate the impact of the aging effects on the integrity of the inaccessible regions in 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.

For those component inspections that do not achieve the inspection coverage criteria stated in the NRC SE (Rev. 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.

Similarly, for those RVI components that are represented by a redundant set of components (e.g., bolting components), the applicant is requested to discuss how the relevant effects of aging will be evaluated in the inaccessible components under two situations: (1) when age-related degradation has been detected in the “sister” components that are accessible to the examination technique, or (2) if accessibility constraints will preclude the applicant from inspecting the recommended minimum number or percentage of components specified in the NRC SE (Rev. 1) on MRP-227. Specifically, for those inspections that achieve the minimum coverage criteria stated in the SE and result in detection of relevant aging effects, the applicant is requested to 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. For those component inspections that do not achieve the inspection coverage criteria stated in the NRC SE (Rev. 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 applicant is also requested to perform an enhancement of the “corrective actions” program element of its RVI Program that will describe the corrective actions that will be implemented if the evaluation of the inaccessible areas cannot demonstrate that the intended function of the impacted RVI components will be maintained during the period of extended operation.

5. Reinspection frequencies: MRP-227-A proposed a 10-year reinspection interval for Westinghouse-design and B&W-design baffle-to-former bolt components and CE-design core shroud bolts, and their associated locking devices, and for those RVI components that are designated as Westinghouse-design, CE-design and B&W-design “Expansion Category” components. The NRC SE (Rev. 1) on MRP-227 identified that further evaluation of the reinspection interval would be necessary only for those applicants or licensees that were proposing to exceed the MRP’s recommended 10-year reinspection interval for the components, as described in Topical Report Condition Item (TRCI) Nos. 5 and 6 in Sections 4.1.5 and 4.1.6 of the NRC SE (Rev. 1) on MRP-227.

   PWR applicants are requested to justify the basis for their reinspection interval if they are proposing to use reinspection intervals for their baffle to former bolts (applicable to Westinghouse and B&W designs) or core shroud bolts (applicable to CE designs), or for their RVI “Expansion Category” components, that are longer than the 10-year reinspection interval proposed in MRP-227-A. Applicants are also requested to identify the increase in the reinspection interval for the applicable components as an exception to the “monitoring and trending” element in GALL AMP XI.M16A. The staff will evaluate such bases on a case by case basis. Otherwise, no further evaluation justifications are necessary if the applicant is proposing to implement the reinspection intervals that are recommended for these components in MRP-227-A.

6. Thermal Aging Embrittlement: MRP-227-A identified that RVI components have the potential to exhibit a decrease in fracture toughness as a result of a neutron irradiation embrittlement mechanism or a thermal aging embrittlement mechanism. The latter is pertinent for those components composed of cast austenitic stainless steel (CASS), martensitic stainless steel (martensitic SS), or precipitation-hardened stainless steel (PH SS) materials.

   In Section 3.3.7 of the NRC SE (Rev. 1) on MRP-227, the staff assessed the impact that thermal aging embrittlement could have on the structural integrity of CASS, PH SS, or martensitic SS RVI components and recommended that PWR applicants for renewal develop a plant-specific analysis to demonstrate that these components will maintain their functions during the period of extended operation (A/LAI No. 7).

   Further evaluation of the thermal aging embrittlement mechanism is necessary for:

   - AMR items in GALL Table IV.B2, IV.B3, or IV.B4 which specifically call out a CASS material and are applicable to the applicant’s IPA, or

   - applicable GALL AMR items with general stainless steel references or plant-specific AMR items, if the applicant’s IPA cannot demonstrate that the stainless steel component is not made from CASS, martensitic SS, or PH SS type materials.

   A/LAI No. 7 states that the plant-specific analysis (1) should consider the impact that thermal aging and neutron irradiation embrittlement mechanisms would have on the fracture toughness properties and structural integrity functions of these components during the period of extended operation, and (2) may need to consider limitations on the accessibility of the components for inspection or on the resolution/sensitivity of the inspection techniques being used to inspect the components to detect cracking. The plant-specific analysis should be consistent with the plant's licensing basis and should address the need to maintain the functionality of the components being evaluated under all licensing basis conditions of operation.

For RVI components that are potentially subject to thermal aging embrittlement, the applicant is requested to discuss and justify: (a) whether a plant-specific analysis will be developed to evaluate the impacts that neutron irradiation embrittlement and thermal aging will have on the fracture toughness properties and structural integrity of the components during the period of extended operation, and if so (b) how the analysis will be used to demonstrate that these components will maintain their intended functions during the period of extended operation. Consistent with A/LAI No. 7, limitations on component accessibility and the ability of the inspection techniques to resolve cracks (i.e., inspection method sensitivity) may be used as part of the justification basis.

7. <u>Use of VT-3 Visual Inspection Techniques for Detection of Cracking</u>: MRP-227-A identifies that VT-3 visual inspections techniques can be used as an acceptable inspection basis for cases where it was demonstrated that a component would be extremely tolerant of a existing flaw in the component, or for a redundant set of components (such as a population or set of redundant bolting components) where the intended function of the overlying assembly would not rely upon and be impacted by a single component failure in the component population.

The "detection of aging effects" recommendation in GALL AMP XI.M16A limits this approach by stating that VT-3 visual methods may only be applied for the detection of cracking in non-redundant RVI components if the flaw tolerance of the component or affected assembly has been demonstrated for easily detected large flaws, even under reduced fracture toughness conditions. For cases involving redundant sets of RVI components (e.g., bolting), ~~the~~ GALL AMP XI.M16A qualifies this by stating that VT-3 methods may be used for the detection of cracking only if the applicant can identify and justify how many of the components could fail while still maintaining the intended function of the relevant RVI assembly.

Consistent with the "detection of aging effects" program element recommendation in GALL AMP XI.M16A, B&W-design, CE-design, and Westinghouse-design applicants are requested to justify the use of VT-3 visual examination techniques for any "Primary Category" RVI components or "Expansion Category" RVI components where VT-3 visual examination techniques are credited as the basis for monitoring cracking in the components.

For non-redundant RVI components fabricated from either CASS, PH SS or martensitic SS, applicants are requested to justify how the performance of the VT-3 examination technique will be capable of detecting a flaw in the components prior to a catastrophic fracture of the components under reduced fracture toughness property conditions induced by a combination of neutron irradiation embrittlement and thermal aging embrittlement. If the IPA determines that the non-redundant components are not sufficiently tolerant of existing flaws or the VT-3 method would not be capable of resolving flaw sizes less than the lower bound limiting critical flaw size (as calculated using NRC-accepted reduced fracture toughness data), the applicant should propose an enhancement of the "detection of aging effects" program element of their RVI Program to propose either EVT-1 or volumetric examinations of the components during the period of extended operation.

For redundant RVI components (e.g., bolting), the applicant should identify, justify, and discuss how many of the components could fail while still maintaining the intended function of the relevant RVI assembly relying upon the integrity of the redundant set of components.

8. <u>Impact of Applicable Technical Specification Requirements or Operating License Requirements on Aging Management Programs for PWR RVI Components</u>: 10 CFR 54.22 requires that each applicant for license renewal identify, as part of its LRA, those technical specification changes (and the justification for the changes) that are necessary to manage the effects of aging during the period of extended operation. Consistent with A/LAI No. 8, Item 4 of the NRC SE (Rev. 1) on MRP-227, each PWR-design license renewal applicant is requested to identify any mandated I&E implementation requirements in the operating license or technical specifications for its facility that are applicable to the RVI components that have been screened in for an AMR. If any identified requirements differ from the related I&E recommendations for the components in MRP-227A, the applicant shall enhance the "scope of program" element of the RVI Program to define these operating license or technical specification requirements for the RVI components. The applicant may submit a license amendment request or technical specifications change request with its LRA in accordance with 10 CFR 54.22, which the staff will review for approval in accordance with the license amendment request acceptance criteria in that are defined and mandated in 10 CFR 50.92.

9. <u>Identification of TLAAs for PWR-Design RVI Components</u>: 10 CFR 54.21(c)(1) requires that each applicant for license renewal provide a list of time-limited aging analyses, as defined in 10 CFR 54.3(a). However, MRP-227-A does not specifically address the resolution of TLAAs that may apply to PWR RVI components.

   Consistent with A/LAI No. 8, Item 5 of the NRC SE (Rev. 1) on MRP-227, PWR applicants for renewal are required by 10 CFR 54.21(c)(1) to identify all analyses in their CLBs that are applicable to in-scope RVI components and meet the definition of a TLAA in 10 CFR 54.3(a). For each such analysis, the applicant shall include an evaluation of the TLAA in its LRA, and shall provide its basis on why the TLAA is acceptable for the period of extended operation, in accordance with 10 CFR 54.21(c)(1). Pursuant to 10 CFR 54.21(c)(2), the applicant shall provide in its LRA a FSAR or UFSAR Supplement summary description for each RVI component analysis in the CLB that conforms to the definition of a TLAA in 10 CFR 54.3.

   For those cumulative usage factor (CUF) analyses that are TLAAs, Section 3.5.1 of the NRC SE (Rev. 1) on MRP-227 states that the PWR Vessel Internals Program may be identified as the basis for demonstrating acceptability of the TLAA in accordance with 10 CFR 54.21(c)(1)(iii) only if the RVI components in the CUF analyses are periodically inspected for fatigue-induced cracking during the period of extended operation. Otherwise, demonstration of acceptability of these TLAAs shall be in accordance with 10 CFR 54.21(c)(1)(i) or (ii), or in accordance with 10 CFR 54.21(c)(1)(iii) using the applicant's program that corresponds to NUREG-1801, Revision 2, AMP X.M1, "Metal Fatigue of Reactor Coolant Pressure Boundary Program." To satisfy the requirements of

ASME Code, Section III, Subsection NG-2160 and NG-3121, the existing fatigue CUF analyses shall include the effects of the reactor coolant system water environment.

Consistent with A/LAI No. 8, Item 5 of the NRC SE (Rev. 1) on MRP-227, the applicant is requested to include in its further evaluation response to SRP-LR Section 3.1.2.2.1, "Cumulative Fatigue Damage," a justification on acceptability of its approach to credit the RVI Program as the basis for accepting the design basis fatigue TLAA for the component in accordance with 10 CFR 54.21(c)(1)(iii), if it chooses to do so. The justification should include a summary on whether MRP-227-A specifies inspection of the RVI component as a "Primary Category" component or "Existing Program Category" component during the period of extended operation.

B. *Further Evaluation "Acceptance Criteria" Recommendations for PWR RVI Components – Recommendations Applicable to Westinghouse-NSSS Designs*

1. Westinghouse-design CRGT support pins: Section 3.2.5.3 of the NRC SE (Rev. 1) on MRP-227 describes further evaluation of the aging management activities for control rod guide tube (CRGT) assembly guide tube support pins (split pins). The NRC SE recommends that the applicant consider activities or actions to: (a) replace its CRGT split pins if the pins are currently fabricated from Alloy X-750 material; or (b) augment its RVI Program to inspect for cracking or loss of material due to wear if the CRGT split pins are fabricated from stainless steel material.

   Consistent with Section 3.2.5.3 of the NRC SE (Rev. 1) on MRP-227, A/LAI No. 3 recommends that the applicant perform an IPA evaluation of its CLB to determine whether the CLB includes any existing NRC-mandated or vendor/supplier-recommended inspection monitoring bases for the CRGT split-pins and whether its split pins are fabricated from either X-750 nickel alloy materials or from stainless steel materials. Based on this evaluation, the applicant should identify the changes to its RVI Program (if any) that need to be made to the component inspection category criteria or replacement schedule criteria for CRGT splits pins, and if appropriate, should identify them as applicable augmented enhancements of the "scope of program," "detection of aging effects," "monitoring and trending," acceptance criteria," and/or "corrective actions" program elements for its RVI Program, particularly if augmentation of the program is warranted. If appropriate, the applicant should justify any enhancements of its program as part its response to these further evaluation recommendations, including proposed replacement schedules for the components. For inspection based programs, the further evaluation response should justify the inspection method(s) selected to detect cracking and/or wear in the components, sample sizes and the baseline and reinspection frequencies that would be applied to the split pin inspections. Refer to GALL AMR items IV.B2.RP-355 and IV.B2.RP-356.

2. <u>Westinghouse-design hold down springs</u>: Section 3.3.5 of Rev. 1 of the NRC SE on MRP-227 recommends further evaluation for the management of loss of preload/loss of compressibility in Westinghouse-design hold down springs made from type 304 austenitic stainless steels.

   For Westinghouse-design facilities with this type of hold-down springs, A/LAI No. 5 of the NRC SE (Rev. 1) on MRP-227 recommends that Westinghouse-design applicants enhance the "detection of aging effects" program element of their RVI Program to identify and define the physical measurement techniques that will be used to monitor loss of preload/loss of compressibility in their RVI hold-down springs and justify the physical measurement technique selected for aging management in their responses to this further evaluation recommendation. The staff also recommends that the Westinghouse-design applicants enhance the "acceptance criteria" program element of their RVI Program to: (1) define the acceptance criteria that will be applied to the physical measurement technique that is selected to manage loss of preload/loss of compressibility in the RVI hold-down springs, and (2) justify their acceptance criteria for the physical measurement technique in their responses to this further evaluation recommendation. Refer to GALL AMR item IV.B2.RP-300.

C. *Further Evaluation "Acceptance Criteria" Recommendations for PWR RVI Components – Recommendations Applicable to Combustion Engineering (CE-NSSS) Designs*

1. <u>CE-design control element assembly (CEA) instrument guide tubes</u>: MRP-227-A identified the CEA instrument guide tubes in the periphery of the CEA as "Primary Category" RVI components and the remaining instrument guide tubes in the CEA as their "Expansion Category" component links. The MRP's augmented inspection basis calls for the performance of a VT-3 visual examination in order to verify that gross cracking has not resulted in missing supports or separation at the welded joints between the tubes and their supports. However, the MRP states that "plant-specific component integrity assessments may be required if degradation is detected and remedial action is needed."

   Consistent with the further evaluation "acceptance criteria" recommendations in SRP-LR Section 3.1.2.2.9.A.7, CE-design applicants for renewal are requested to justify the use of VT-3 examination techniques for the detection of cracking in their CEA instrument guide tube components. Refer to GALL AMR items IV.B3.RP-312 and IV.B3.RP-313.

2. <u>Welded CE-design core shrouds</u>: Section 3.3.5 of the NRC SE (Rev. 1) on MRP-227, states that MRP-227-A indicated that physical measurement methods would need to be performed in order to monitor for changes in dimension/distortion in welded CE-design core shroud designs containing two core shroud segments (shell courses), but the specific physical measurement techniques for performing verification of changes in dimension/distortion in the gap between the upper and lower core shroud segments were not within the scope of the MRP-227-A. Further, MRP-227-A did not include appropriate acceptance criteria for the physical measurement techniques that would be applied to gap areas in these types of core shroud designs.

   A/LAI No. 5 of the NRC SE (Rev. 1) on MRP-227 requested that CE applicants owning these types of core shroud designs should define the physical measurement methods, and the acceptance criteria for these methods, that would be applied to the inspection and evaluation of the gap areas in the welded, two-segment core shroud designs.

   For CE-designed plants with this type of core shroud design, the applicants are requested to enhance the RVI Program consistent with the recommendation in A/LAI No. 5. The "detection of aging effects" program element of their RVI Program should identify and define the physical measurement technique and inspection frequency that will be used to monitor for changes in dimension/distortion in the gap areas of their core shroud assembly structures during the period of extended operation, and justify the use of the physical measurement technique selected for inspection. The "acceptance criteria" program element of their RVI Program should define the acceptance criteria that will be applied to these physical measurement technique(s), and should justify these acceptance criteria for these techniques. The information and justification bases for enhancing these two program elements should be provided in the further evaluation section of the LRAs that addresses aging management of RVI components. Refer to GALL AMR item IV.B3.RP-326.

3. <u>CE-design lower core flange welds, core support plates, and fuel alignment plates</u>: As endorsed in the NRC SE (Rev. 1) on MRP-227, MRP-227-A identified that "Primary Category" 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 in accordance with 10 CFR Part 54.21(c)(1)(i) or (ii) during the period of extended operation. These components are the CE-design lower core flange welds in the core support barrel assembly, the core support plates in the lower core support assembly (applicable to CE-designed reactors that include a core support plate), and the fuel alignment plates in those CE-designed reactors that are designed with full height shroud plates.

   Therefore, further evaluation of the components is necessary to determine whether the components have been analyzed with a design basis fatigue analysis that conforms to the definition of a TLAA in 10 CFR 54.3(a), and if so, whether the TLAA can demonstrate that fatigue-induced cracking will not initiate in the components during the period of extended operation (i.e., acceptance of the TLAAs in accordance with 10 CFR 54.21(c)(1)(i) or (ii)), or else that cumulative fatigue damage or fatigue-induced cracking will be adequately managed during the period of extended operation in accordance with 10 CFR 54.21(c)(1)(iii).

   Applicants are requested to assess whether their CLB includes applicable design basis fatigue TLAAs for these CE-design RVI components and whether the AMR items on cumulative fatigue damage in GALL AMRs IV.B3.RP-389, IV.B3.RP-390, and IV.B3.RP-391 are applicable to their LRA. If the CLB includes applicable design basis fatigue analyses for these components and the analyses are determined to conform to the definition of a TLAA in 10 CFR 54.3, the applicant should provide its basis for dispositioning the TLAAs in accordance with 10 CFR 54.21(c)(1).

   If 10 CFR 54.21(c)(1)(iii) is used to disposition the TLAA and the applicant's cycle counting activity of its AMP corresponding to GALL AMP X.M1, "Metal Fatigue of Reactor Coolant Pressure Boundary" is the basis for accepting these TLAAs, the staff requests that the applicant provide in its LRA: (1) a sufficient discussion on how the AMP's cycle counting activity "corrective actions" bases would result in the implementation of the MRP's recommended EVT-1 inspection bases and criteria for the components, as given in Tables 4-2 and 5-2 of MRP-227-A; and (2) a justification of the basis for crediting the cycle counting activities of the Fatigue Monitoring Program (or its equivalent) as the basis for accepting the design basis fatigue analyses (e.g., CUF or I_t type of fatigue analyses) and managing cumulative fatigue damage or fatigue-induced cracking in lieu of the MRP's recommended EVT-1 visual criteria that would be used to inspect the components under the applicant's RVI Program. Use GALL AMR Items IV.B3.RP-389, IV.B3.RP-390, and IV.B3.RP-391 if design basis fatigue TLAAs will be relied upon as the basis for managing cumulative fatigue damage in the components or for concluding that fatigue induced cracking will not occur in the components during the period of extended operation.

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 I_t fatigue analyses for these components; (2) the IPA identifies that SCC, IASCC, or intergranular attack are plausible plant-specific mechanisms for inducing cracking in the components; or (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). Use GALL AMR items IV.B3.RP-333, IV.B3.RP-338, or IV.B3.RP-343 if the MRP's recommended EVT-1 will be used as the aging management basis for the managing cracking in the components, or in addition to the applicable TLAA AMR item (i.e., in addition to AMR items IV.B3.RP-389, IV.B3.RP-390, or IV.B3.RP-391) if the applicant's RVI Program and the MRP's recommended EVT-1 basis will be used for accepting the design basis fatigue TLAA in accordance with10 CFR 54.21(c)(1)(iii).

4. <u>CE-design fuel alignment pins</u>: Section 3.2.5.3 of the NRC SE (Rev. 1) on MRP-227, (A/LAI No. 3) specifies further evaluation to verify whether the VT-3 examinations performed on CE-design fuel alignment pins under the ASME Section XI "Existing Program" requirements will be adequate to manage cracking and loss of fracture toughness in the components, or else to determine whether the program's monitoring bases for these components need to be enhanced to specify implementation of more conservative inspection techniques, such as VT-1, EVT-1, or volumetric inspection techniques.

   CE-design applicant's for renewal are requested to justify the use of VT-3 examinations of their fuel alignment pins consistent with Section 3.2.5.3 of the NRC SE (Rev. 1) on MRP-227, and the criteria in A/LAI No. 3 for these components. This evaluation can be implemented in conjunction with the further evaluation "acceptance criteria" recommendations in SRP-LR Subsection 3.1.2.2.9.A.7 on acceptability of VT-3 visual examination methods may be used for the monitoring and detection of cracking in a component. Refer to GALL AMR items IV.B3.RP-334 and IV.B3.RP-364.

5. <u>CE-design incore instrumentation (ICI) flux thimble tubes and thermal shield positioning pins</u>: MRP-227-A identified that the monitoring programs for CE-design ICI flux thimble tubes and thermal shield positioning pins were limited to plant-specific recommendations that had no generic reference. The MRP recommended that CE-design applicants should review their specific design information, upgrade status reports, and plant commitments for these components in order to verify whether or not their CLB or current design basis had applicable existing condition monitoring programs for components in place.

   In Section 3.2.5.3 of the NRC SE (Rev. 1) on MRP-227 and A/LAI No. 3, the staff requested that CE-design applicants perform a review of their CLB to identify the "Existing Program" criteria for their ICI flux thimble tubes and thermal shield positioning pins and to provide an evaluation of the existing program bases for these components. A/LAI No. 3 states that the evaluation should also include an appropriate justification to: (1) verify the acceptability of their existing monitoring programs for these components, or (2) to identify the needed changes to the program that should be implemented to manage the aging of these components during the period of extended operation.

   CE-design applicants for renewal are requested to justify the aging management bases for their ICI flux thimble tubes and thermal shield positioning pins relative to the criteria in

A/LAI No. 3 and the further evaluation criteria for these components stated in the previous paragraph. Based on this evaluation, the applicant should identify those changes that need to be made to the component inspection category criteria for their incore instrumentation thimble tubes and thermal shield positioning pins (if any) as part of applicable augmented enhancements of the "scope of program," "detection of aging effects," "monitoring and trending," acceptance criteria," and/or "corrective actions" program elements for its RVI Program, as appropriate. The applicant should justify any enhancements of its program as part its response to these further evaluation recommendations. Otherwise, CE-design applicants for renewal are requested to provide appropriate justifications if inspections of the ICI flux thimble tubes or thermal shield positioning pins will not be performed as part of their aging management programs. Refer to GALL AMR items IV.B3.RP-357 and IV.B3.RP-400.

D. *Further Evaluation "Acceptance Criteria" Recommendations for PWR RVI Components – Recommendations Applicable to Babcock and Wilcox NSSS (B&W-NSSS) Designs*

1. B&W-design core support shield assembly upper flange welds: MRP-227-A identified that further evaluation and augmented aging management of cracking and loss of fracture toughness would be necessary only if the IPA confirms that the welds were not subjected to a post-weld heat treatment/stress relief process during initial fabrication of the welds. For cases where the upper flange welds have not been post-weld heat treated, the MRP recommended inspections of the B&W upper flange welds that are analogous to MRP's recommended criteria for inspecting corresponding core support structure upper flange welds in Westinghouse-designed and CE-designed units. The MRP's recommended inspections call for the Westinghouse or CE applicants to perform enhanced visual (EVT-1) examinations of their analogous upper flange weld components no later than two (2) refueling outages from the beginning of the license renewal period, and for subsequent EVT-1 re-examinations to be performed on a 10-year re-inspection interval.

   To address Section 3.2.5.4 of the NRC SE (Rev. 1) on MRP-227 and A/LAI No. 4, applicants with B&W-design RVI components are requested to perform a review of their CLB to determine whether the upper flange weld in their core support shield assemblies were subjected to a post-weld heat treatment/stress relief process during initial fabrication of the welds. If the applicant confirms that the welds were appropriately stress--relieved, the applicant should identify the stress relief heat treatment process for the upper flange welds as a plant-specific enhancement of the "preventative actions" program element for their RVI Program.

   For those applicants with B&W-design RVI components that cannot confirm appropriate stress relief of the welds during fabrication, the staff recommends that the applicant provide appropriate plant-specific enhancements of the "scope of program," "detection of aging effects," "monitoring and trending," and "acceptance criteria" program elements that establish how the program will be adjusted to inspect these upper flange welds consistent with the MRP's bases in MRP-227-A Tables 4-3 and 5-3 for inspecting, monitoring, and accepting potential relevant indications in analogous Westinghouse-designed upper core barrel flange welds, or in MRP-227-A Tables 4-2 and 5-2 for analogous CE-designed core support barrel upper flange welds. The staff also recommends that the applicant propose inspection coverage criteria for the examinations that are consistent with the staff's examination coverage criteria that are provided in Sections 3.3.1 and 4.3.1 of the NRC SE (Rev. 1) on MRP-227, and propose applicable "Expansion Category" components for the core support structure upper flange weld examinations that are similar to the “Expansion Category” component links in MRP-227-A Table 4-3 for analogous Westinghouse-designed core barrel upper flange welds, or in MRP-227-A Table 4-2 for analogous CE-designed core support barrel upper flange welds. Otherwise, the applicants should justify inspection bases that differ from these further evaluation recommendations.

The staff also recommends that applicants apply GALL AMR items IV.B4.RP-400 and IV.B4.RP-401 as their “Table 2” AMR bases for managing cracking due to SCC and loss of fracture toughness due to neutron irradiation embrittlement in the core support structure upper flange weld, with the added recommendation that applicants include a plant-specific footnote on their AMR items that defines the type of aging management basis that will be used for the upper flange weld (i.e., AMP preventative action through the weld fabrication stress relief process, or inspection of the welds).

2. <u>B&W-design Expansion Category components</u>: Further evaluation of the following “Expansion Category” RVI components in the core barrel assembly of B&W-designed reactors is recommended: (1) internal baffle-to-baffle bolts; (2) external baffle-to-baffle bolts and their locking devices; (3) core barrel-to-former bolts and their locking devices; (4) former plates; and (5) core barrel cylinder circumferential welds (girth welds) and vertical seam welds (axial welds). Specifically, MRP-227-A identified that the external baffle-to-baffle bolts and their locking devices, core barrel-to-former bolts and their locking devices, former plates, and core barrel cylinder circumferential and axial welds are inaccessible to examination. The MRP also identified that the industry has yet to define inspection techniques for the internal baffle-to-baffle bolts in B&W-designed reactors, even though the bolts are accessible for inspection.

   Section 3.3.6 of the NRC SE (Rev. 1) on MRP-227 and A/LAI No. 6 requested that applicants with B&W-design reactors describe the type of engineering analysis that would be performed to demonstrate that the integrity and functionality of these “Expansion Category” components, and of the RVI assemblies containing the components, will be maintained during the period of extended operation if age-related degradation is detected in any of the related “Primary Category” components. Consistent with A/LAI No. 6, the applicant is requested to describe the type of engineering analysis that would be performed to demonstrate that the integrity and functionality of the following “Expansion Category” components during the period of extended operation: (1) internal baffle-to-baffle bolts; (2) external baffle-to-baffle bolts and their locking devices; (3) core barrel-to-former bolts and their locking devices; (4) former plates; and (5) core barrel cylinder circumferential and axial welds. The applicant is also requested to identify this analysis as an applicable plant-specific enhancement of the “monitoring and trending” program element for its RVI Program.

   Otherwise, the staff recommends that applicants with B&W-design RVI plan to replace core barrel assembly components as an alternative to the recommended analysis or if age-related degradation is detected in the “Primary Category” components and the recommended ensuing analysis could not ensure the integrity and functionality of these “Expansion Category” components during the period of extended operation. If a replacement schedule is selected as the appropriate basis for aging management, the staff recommends that the applicant should include the replacement schedule basis as a plant-specific enhancement of the RVI Program. Applicants with B&W-design RVI should also justify its basis for accepting the integrity and functionality of the applicable inaccessible components during the period of extended operation as part of its response to this further evaluation recommendation. Refer to AMR items IV.B4.RP-243, IV.B4.RP-243a, IV.B4.RP-244, IV.B4.RP-244a, IV.B4.RP-250, IV.B4.RP-250a, IV.B4.RP-375 and IV.B4.RP-375a.

3.1.2.2.10 Loss of Fracture Toughness due to Neutron Irradiation Embrittlement, Changes in Dimension due to Void Swelling, Loss of Preload due to Stress Relaxation, or Loss of Material due to Wear

The previous "acceptance criteria" in SRP-LR Sections 3.1.2.2.10 for managing loss of fracture toughness, loss of preload, changes in dimension, or loss of material in the inaccessible areas of partially accessible components have been incorporated into and are replaced by the further evaluation "acceptance criteria" recommendations in SRP-LR Section 3.1.2.2.9.A, Part 6 (3.1.2.2.9.A.6).

3.1.2.2.12 Cracking due to Fatigue

The previous "acceptance criteria" in SRP-LR Sections 3.1.2.2.12 for managing fatigue-induced cracking in the CE-design core barrel assembly lower flange welds, fuel alignments plates, and lower core structure support plates have been incorporated into and are replaced by the further evaluation "acceptance criteria" recommendations in SRP-LR Section 3.1.2.2.9.C, Part 3 (3.1.2.2.9.C.3).

3.1.2.2.13 Cracking due to Stress-Corrosion Cracking and Fatigue

The previous "acceptance criteria" in SRP-LR Sections 3.1.2.2.13 for managing fatigue-induced, SCC-induced, and IASCC-induced cracking in Westinghouse-design control rod guide tube support pins (split pins) have been incorporated into and are replaced by the further evaluation "acceptance criteria" recommendations in SRP-LR Section 3.1.2.2.9.B, Part 1 (3.1.2.2.9.B.1).

3.1.2.2.14 Loss of Material due to Wear

The previous "acceptance criteria" in SRP-LR Sections 3.1.2.2.14 for managing loss of materials due to wear in Westinghouse-design control rod guide tube support pins (split pins) and CE-design incore instrumentation flux thimble tubes have been incorporated into and are replaced by the further evaluation "acceptance criteria" recommendations in SRP-LR Section 3.1.2.2.9.B, Part 1 (3.1.2.2.9.B.1) for managing loss of material due wear in the Westinghouse-design control rod guide tube support pins (split pins) and in SRP-LR Section 3.1.2.2.9.C, Part 5 (3.1.2.2.9.C.5) for managing loss of material due to wear in CE-design incore assembly flux thimble tubes.

Updated Draft Further Evaluation "Review Procedure" Sections for PWR RVI Components in SRP-LR Section 3.1.3.2

REVIEW PROCEDURES

3.1.3.2.9 Augmented Inspection Bases for PWR Vessel Internal Components

The MRP's augmented inspection program for PWR RVI components is an NRC-endorsed and industry-defined AMP for PWR-designed light water reactors. However, the MRP program includes both MRP-defined inspection and evaluation activities and those plant-specific inspection or evaluation activities that an applicant for renewal may need to define for its RVI Program. The NRC has reflected this type of basis in GALL AMP XI.M16A, "PWR Vessel Internals." The NRC has issued specific A/LAIs in the NRC SE (Rev. 1) on MRP-227 to assist a PWR applicant for renewal in identifying those inspection-based or evaluation-based activities that an applicant would need to define as plant-specific aspects of its RVI Program, and has reflected these A/LAIs, or specific GALL AMP XI.M16A guidance criteria, in its further evaluation "acceptance criteria" recommendations for PWR RVI components, as given in Section 3.1.2.2.9 of the SRP-LR and its subsections.

PWR applicants for license renewal should use their IPA process as the basis for: (1) responding to the A/LAIs on the MRP-227 methodology and the further evaluation "acceptance criteria" recommendations in SRP-LR Section 3.1.2.2.9 that are applicable to their facilities, (2) identifying those additional plant-specific Table 2 AMR items for their RVI components that need to included in their LRAs, and (3) identifying those plant-specific inspection or evaluation criteria that need to be defined as plant-specific enhancements for their RVI Program.

The NRC reviewer performs its review of a PWR applicant's further evaluation response bases for its RVI components based on a comparison against the evaluation criteria in the following NRC guidance documents: (1) the SRP-LR further evaluation "acceptance criteria" recommendations for PWR RVI components in Section 3.1.2.2.9 of the SRP-LR; (2) the A/LAIs that are applicable to the applicant's RVI components in the NRC SE (Rev. 1) on MRP-227; (3) Table 1 AMR items for the PWR RVI components in SRP-LR Table 3.1-1; (4) Table 2 AMR items in GALL Table IV.B2, IV.B3, or IV.B4 that are applicable to the plant's NSSS design; and (5) the program element criteria that are given in GALL AMP XI.M16A, "PWR Vessel Internals." The NRC reviewer performs a review of any plant-specific AMR items for the RVI components on a case-by-case basis.

The reviewer uses relevant information in the applicant's CLB. The following sources apply: (1) any relevant requirements that may apply to the RVI components in the plant operating license or technical specifications; (2) the safety analysis report for the facility (e.g., FSAR, USAR, UFSAR); (3) ASME Section III design requirements and Section XI Inservice Inspection requirements for those RVI components defined as ASME core support structure components; and (4) applicable NRC generic communications, and any applicant-made reply bases and commitments made in response to those communications (applicable only to generic communications requiring an applicant or licensee response).

3.1.3.2.10 Loss of Fracture Toughness due to Neutron Irradiation Embrittlement, Changes in Dimension due to Void Swelling, Loss of Preload due to Stress Relaxation, or Loss of Material due to Wear

The previous “review procedures” in SRP-LR Sections 3.1.3.2.10 for managing loss of fracture toughness, loss of preload, changes in dimension, or loss of material in the inaccessible areas of partially accessible components have been incorporated into and are replaced by the further evaluation “review procedure” recommendations in SRP-LR Section 3.1.3.2.9.

3.1.3.2.12 Cracking due to Fatigue

The previous “review procedures” in SRP-LR Sections 3.1.3.2.12 for evaluating fatigue-induced cracking in the CE-design core barrel assembly lower flange welds, fuel alignments plates, and lower core structure support plates have been incorporated into and are replaced by the further evaluation “review procedure” recommendations in SRP-LR Section 3.1.3.2.9.

3.1.3.2.13 Cracking due to Stress-Corrosion Cracking and Fatigue

The previous “review procedures” in SRP-LR Sections 3.1.3.2.13 for evaluating fatigue-induced, SCC-induced, and IASCC-induced cracking in Westinghouse-design control rod guide tube support pins (split pins) have been incorporated into and are replaced by the further evaluation “review procedure” recommendations in SRP-LR Section 3.1.3.2.9.

3.1.3.2.14 Loss of Material due to Wear

The previous “review procedures” in SRP-LR Sections 3.1.3.2.14 for evaluating loss of material due to wear in Westinghouse-design control rod guide tube support pins (split pins) and CE-design incore instrumentation flux thimble tubes have been incorporated into and are replaced by the further evaluation “review procedure” recommendations in SRP-LR Section 3.1.3.2.9.

Proposed 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 (I&E) guidelines in EPRI Report No. ~~1016596~~1022863 (MRP-227-A) and EPRI 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 SCC, 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; (d) dimensional changes and potential loss of fracture toughness due to void swelling and irradiation growth; and (e) loss of preload due to thermal and irradiation-enhanced stress relaxation or creep. The program includes applicant defined plant-specific I&E criteria that are established in response to applicable Applicant/Licensee Action Items (A/LAIs) on the MRP-227-A methodology 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).	Program should be implemented prior to period of extended operation	GALL IV / SRP 3.1

Appendix A, Section 3 – Proposed Changes to "Table 1" AMR Items for PWR Vessel Internals Table 3.1-1 of NUREG-1800, Revision 2 (Proposed Changes to SRP-LR Table 3.1-1

Table 3.1-1 Summary of Aging Management Programsfor 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 SRP-LR Section ~~subsection~~ 3.1.2.2.1 as performed in conjunction with SRP-LR Section ~~subsection~~ 3.1.2.2.9.A.9 for identifying CUF or I_t fatigue TLAAs for PWR RVI components.	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 (B&W)	Stainless steel (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 SRP-LR Section ~~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, and fatigue	Chapter XI.M16A, "PWR Vessel Internals"	Yes, evaluate inaccessible areas for impact on intended functions if accessible Primary, Expansion or Existing Program components indicate aging effects that need management (See	IV.B2.RP-268 IV.B3.RP-309 IV.B4.RP-238	N/A N/A N/A

Table 3.1-1 Summary of Aging Management Programsfor 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**
					SRP-LR Section ~~subsection~~ 3.1.2.2.9.A.4)		
24	PWR	Stainless steel (including CASS, martensitic SS, and PH SS) and 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 and for CASS, martensitic SS, or PH SS due to thermal aging embrittlement; or changes in dimension 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"	Yes, evaluate inaccessible areas for impact on intended functions if accessible Primary, Expansion or Existing Program components indicate aging effects that need management (See SRP-LR Section ~~subsection~~ ~~3.1.2.2.10~~ 3.1.2.2.9.A.4)	IV.B2.RP-269 IV.B3.RP-311 IV.B4.RP-239	N/A N/A N/A
26	PWR (CE)	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 CE plants with core shrouds assembled with full height shroud plates) exposed to reactor coolant and neutron flux;	Cracking due to fatigue; loss of fracture toughness due to neutron irradiation embrittlement (for core support plates)	Chapter XI.M16A, "PWR Vessel Internals," (and additionally Chapter XI.M2, "Water Chemistry," if there is a plant-specific basis for concluding cracking may also initiate by either stress-corrosion cracking, irradiation-assisted stress-corrosion cracking or intergranular attack mechanisms) ~~if fatigue life cannot be confirmed by TLAA~~	Yes, evaluate to determine whether cracking due to fatigue can be adequately managed by a fatigue analysis TLAA ~~the potential locations and extent of cracking~~ (See SRP-LR Section ~~subsection~~ ~~3.1.2.2.12~~ 3.1.2.2.9.C.3) Additional generic further evaluation	Primary Category IV.B3.RP-333 IV.B3.RP-338 IV.B3.RP-343 IV.B3.RP-365 (i.e., primary category AMRs if management of cracking initiated by a fatigue mechanism	 N/A N/A N/A N/A

Table 3.1-1 Summary of Aging Management Programsfor 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 Lower support structure: core support plate (applicable to CE plants designed with a core plate) exposed to reactor coolant and neutron flux			recommendations may apply on an IPA-specific basis. See Items 23 and 24 and SRP-LR Section 3.1.2.2.9.A.4 for evaluating inaccessible areas in partially accessible components. See SRP-LR Section 3.1.2.2.9.A.6 for evaluating the impact of thermal aging embrittlement on the fracture toughness property of the material if the IPA confirms the components are made from CASS, PH SS, or martensitic SS materials.	cannot be demonstrated by a TLAA or if IPA independently identifies that SSC, IASCC, or IGA are plausible crack initiation mechanisms) (See AMR Items IV.B3.RP-389, IV.B3.RP-390, and IV.B3.RP-391 for the corresponding GALL TLAA references under Item 3)	
27	PWR (West.)	~~Nickel alloy~~ Westinghouse control rod guide tube (CRGT) assemblies~~,~~: nickel alloy guide tube support pins (split pins) exposed to reactor coolant and neutron flux	Cracking due to stress-corrosion cracking and fatigue and 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~~ – Further evaluation of Westinghouse-design CRGT support pins (split pins) is recommended. (See SRP-LR Section ~~subsection~~ ~~3.1.2.2.13~~ 3.1.2.2.9.B.1)	Existing Program IV.B2.RP-355 IV.B2.RP-356	 N/A N/A

Table 3.1-1 Summary of Aging Management Programsfor 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**
27a	PWR (West)	Westinghouse alignment and interfacing components: austenitic stainless steel hold down springs exposed to reactor coolant and neutron flux	Loss of preload due to thermal and irradiation enhanced stress relaxation; loss of material due to wear	Chapter XI.M16A, "PWR Vessel Internals"	Yes, further evaluation of the physical measurement monitoring basis is necessary (See SRP-LR Section 3.1.2.2.9.B.2)	Primary Category IV.B2.RP-300	N/A
28	PWR (CE)	~~Nickel alloy Westinghouse control rod guide tube (CRGT) assemblies,: guide tube support pins exposed to reactor coolant and neutron flux; and~~ Zircaloy-4 Combustion Engineering incore instrumentation (ICI): thimble tubes exposed to reactor coolant and neutron flux; and Stainless steel Combustion Engineering Core support barrel assembly: thermal shield positioning pins 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	~~A plant-specific aging management program is to be evaluated~~Chapter XI.M16A, "PWR Vessel Internals"	Yes, ~~plant-specific~~ further evaluation of CE-design ICI thimble tubes and thermal shield positioning pins is recommended (See SRP-LR Sections ~~subsection~~ ~~3.1.2.2.14~~ 3.1.2.2.9.C.5) Additional generic further evaluation recommendations may apply on an IPA-specific basis. See Items 23 and 24 and SRP-LR Section 3.1.2.2.9.A.4 for evaluating inaccessible areas in partially accessible components. See SRP-LR Section 3.1.2.2.9.A.7 if crediting VT-3 visual techniques for the detection of cracking.	Existing Program ~~IV.B2.RP-356~~ IV.B3.RP-357 IV.B3.RP-400	~~N/A~~ N/A N/A

Table 3.1-1 Summary of Aging Management Programsfor 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**
28a	PWR (CE)	Combustion Engineering Lower support structure: stainless steel (A-286 is normal type) fuel alignment pins 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 creep; or loss of material due to wear; or cracking due to stress-corrosion cracking	Chapter XI.M16A, "PWR Vessel Internals"	Yes, further evaluation of CE-design fuel alignment pins is recommended (See SRP-LR Section 3.1.2.2.9.C.4) Additional generic further evaluation recommendations may apply on an IPA-specific basis. See Items 23 and 24 and SRP-LR Section 3.1.2.2.9.A.4 for evaluating inaccessible areas in partially accessible components. See SRP-LR Section 3.1.2.2.9.A.6 for evaluating the impact of thermal aging embrittlement on the fracture toughness property of the material if the IPA confirms the components are made from CASS, PH SS, or martensitic SS materials.	Existing Program IV.B3.RP-334 IV.B3.RP-336	IV.B3-23 (R-167) IV.B3-22 (R-170)

Table 3.1-1 Summary of Aging Management Programsfor 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**
28b	PWR (CE)	Combustion Engineering control element assemblies (CEA): stainless steel or nickel alloy instrument guide tubes exposed to reactor coolant and neutron flux	Cracking due to stress-corrosion cracking or fatigue	Chapter XI.M2, "Water Chemistry," and Chapter XI.M16A, "PWR Vessel Internals."	Yes, further evaluation of CEA instrument guide tubes is recommended (See SRP-LR Section 3.1.2.2.9.C.1). Additional generic further evaluation recommendations may apply on an IPA-specific basis. See Items 23 and 24 and SRP-LR Section 3.1.2.2.9.A.4 for evaluating inaccessible areas in partially accessible components. See SRP-LR Section 3.1.2.2.9.A.5 if proposing reinspection frequencies for CE Expansion Category components that are in excess of the MRP recommended frequency of once every ten years.	Primary Category IV.B3.RP-312 Expansion Category IV.B3.RP-313	IV.B3-2 (R-149) N/A
28c	PWR (CE)	Combustion Engineering core shroud assemblies (applicable to welded core shroud designs assembled in vertical	Cracking due to stress-corrosion cracking or fatigue; changes in dimension due to distortion or	Chapter XI.M16A, "PWR Vessel Internals."	Yes, further evaluation is recommended to define the physical measurement method that will be used to	Primary Category IV.B3.RP-326 IV.B3.RP-326a	N/A N/A

Table 3.1-1 Summary of Aging Management Programsfor 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**
		segments); stainless steel core shrouds exposed to reactor coolant and neutron flux	void swelling; loss of fracture toughness due to neutron irradiation embrittlement		inspect the gap area in these types of core shroud designs (See SRP-LR Section 3.1.2.2.9.C.2). Additional generic further evaluation recommendations may apply on an IPA-specific basis. See Item 24 and SRP-LR Section 3.1.2.2.9.A.4 for evaluating inaccessible areas in partially accessible components. See SRP-LR Section 3.1.2.2.9.A.6 for evaluating the impact of thermal aging embrittlement on the fracture toughness property of the material if the IPA confirms the components are made from CASS, PH SS, or martensitic SS materials.		

Table 3.1-1 Summary of Aging Management Programsfor 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 components, 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”	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 (B&W)	Stainless steel or nickel alloy Babcock and 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 component-specific further evaluation recommendations for the components within this commodity group. However, generic further evaluation recommendations may apply. See Item 23 and SRP-LR Section 3.1.2.2.9.A.4 for evaluating inaccessible areas in partially accessible components. See SRP-LR Section 3.1.2.2.9.A.5 if proposing reinspection frequencies for B&W Expansion Category components, or for B&W baffle-to-former bolts and their locking devices, that are in excess of the MRP recommended	~~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~~ No Additional Measures IV.B4.RP-236 Primary Category IV.B4.RP-242a IV.B4.RP-241 IV.B4.RP-241a IV.B4.RP-247 IV.B4.RP-247a IV.B4.RP-248 IV.B4.RP-248a IV.B4.RP-252a IV.B4.RP-256 IV.B4.RP-256a	~~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~~ N/A N/A IV.B4-7 (R-125) N/A IV.B4-13 (R-194) N/A IV.B4-25 (R-210) N/A N/A IV.B4-25 (R-210) N/A

Table 3.1-1 Summary of Aging Management Programsfor 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**
					frequency of once every ten years. See SRP-LR Section 3.1.2.2.9.A.7 if crediting VT-3 visual methods for the detection of cracking.	IV.B4.RP-258a IV.B4.RP-259a IV.B4.RP-261 Expansion Category 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	N/A N/A IV.B4-32 (R-203) 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
52	PWR (CE)	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 component-specific further evaluation recommendations for the components within this commodity group. However, generic further evaluation recommendations may apply. See Item 23 and SRP-LR Section 3.1.2.2.9.A.4 for evaluating inaccessible areas in partially accessible components.	~~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.B3-12 (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 Programsfor 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**
					See SRP-LR Section 3.1.2.2.9.A.5 if proposing reinspection frequencies for CE Expansion Category components, or for CE core shroud bolts, that are in excess of the MRP recommended frequency of once every ten years. See SRP-LR Section 3.1.2.2.9.A.7 if crediting VT-3 visual methods for the detection of cracking.	No Additional Measures IV.B3.RP-306 Primary Category IV.B3.RP-314 IV.B3.RP-322 IV.B3.RP-324 IV.B3.RP-326a IV.B3.RP-327 IV.B3.RP-342 IV.B3.RP-358 IV.B3.RP-362a IV.B3.RP-363 Expansion Category IV.B3.RP-316 IV.B3.RP-323 IV.B3.RP-325 IV.B3.RP-327a IV.B3.RP-329 IV.B3.RP-330 IV.B3.RP-362c Existing Program IV.B3.RP-320	 N/A IV.B3-9 (R-162) N/A N/A N/A IV.B3-15 (R-155) N/A N/A N/A N/A IV.B3-9 (R-162) N/A N/A N/A IV.B3-12 (R-155) IV.B3-23 (R-167) N/A IV.B3-9 (R-162)

Table 3.1-1 Summary of Aging Management Programsfor 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 (West.)	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 component-specific further evaluation recommendations for the components within this commodity group. However, generic further evaluation recommendations may apply. See Item 23 and SRP-LR Section 3.1.2.2.9.A.4 for evaluating inaccessible areas in partially accessible components. See SRP-LR Section 3.1.2.2.9.A.5 if proposing reinspection frequencies for Westinghouse Expansion Category components, or for Westinghouse baffle-to-former bolts, that are in excess of the MRP recommended frequency of once every ten years. See SRP-LR Section 3.1.2.2.9.A.7 if crediting VT-3 visual methods for the	~~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~~ No Additional Measures IV.B2.RP-265 Primary Category 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-387 Expansion Category IV.B2.RP-273 IV.B2.RP-278	~~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~~ N/A 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 IV.B2-10 (R-125) IV.B2-8 (R-120)

Table 3.1-1 Summary of Aging Management Programsfor 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**
					detection of cracking.	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 Existing Program IV.B2.RP-289 IV.B2.RP-301 IV.B2.RP-345 IV.B2.RP-346 IV.B2.RP-399	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 IV.B2-20 (R-130) IV.B2-40 (R-112) N/A N/A N/A
54	PWR (West.)	Stainless steel bottom mounted instrument system flux thimble tubes (with or without chrome plating) exposed to reactor coolant and neutron flux	Loss of material due to wear	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 (West.)	Stainless steel core 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~~ See SRP-LR Section 3.1.2.2.9.A.7 if crediting VT-3 visual techniques for the detection of cracking. See Items 23 and 24, and SRP-LR Section 3.1.2.2.9.A.4 for evaluating inaccessible areas in partially accessible components	Primary Category IV.B2.RP-302	 N/A

Table 3.1-1 Summary of Aging Management Programsfor 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
56	PWR (CE)	Stainless steel (SS, including CASS, PH SS or martensitic SS) or nickel alloy Combustion Engineering 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 dimension due to void swelling; 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 component-specific further evaluation recommendations for the components within this commodity group. However, generic further evaluation recommendations may apply. See Item 24 and SRP-LR Section 3.1.2.2.9.A.4 for evaluating inaccessible areas in partially accessible components. See SRP-LR Section 3.1.2.2.9.A.5 if proposing reinspection frequencies for CE Expansion Category components, or for CE core shroud bolts, that are in excess of the MRP recommended frequency of once every ten years. See SRP-LR Section 3.1.2.2.9.A.6 for evaluating the impact of thermal aging embrittlement on the fracture toughness	~~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~~ ~~IV.B3.RP-366~~ No Additional Measures IV.B3.RP-307 Primary Category 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 Expansion Category IV.B3.RP-317 IV.B3.RP-331 IV.B3.RP-359a	~~N/A~~ ~~IV.B3-7 (R-165)~~ ~~IV.B3-7 (R-165)~~ ~~IV.B3-8 (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~~ ~~N/A~~ N/A IV.B3-7 (R-165) IV.B3-8 (R-163) N/A N/A N/A N/A N/A IV.B3-7 (R-165) N/A N/A

Table 3.1-1 Summary of Aging Management Programsfor 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**
					property of the material if the IPA confirms the components are made from CASS, PH SS, or martensitic SS materials.	IV.B3.RP-361 Existing Program IV.B3.RP-319 IV.B3.RP-332	N/A IV.B3-9 (R-162) IV.B3-17 (R-156)
58	PWR (B&W)	Stainless steel (SS, including CASS, PH SS or martensitic SS) or nickel alloy Babcock and Wilcox 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 dimension due to void swelling; 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," and Chapter XI.M2, "Water Chemistry"	No component-specific further evaluation recommendations for the components within this commodity group. However, generic further evaluation recommendations may apply. See Item 24 and SRP-LR Section 3.1.2.2.9.A.4 for evaluating inaccessible areas in partially accessible components. See SRP-LR Section 3.1.2.2.9.A.5 if proposing reinspection frequencies for B&W Expansion Category components, or for B&W baffle-to-former bolts and their locking devices, that are in	~~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~~ No Additional Measures IV.B4.RP-237 Primary Category 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	~~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)~~ N/A 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 Programsfor 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**
					excess of the MRP recommended frequency of once every ten years. See SRP-LR Section 3.1.2.2.9.A.6 for evaluating the impact of thermal aging embrittlement on the fracture toughness property of the material if the IPA confirms the components are made from CASS, PH SS, or martensitic SS materials.	IV.B4.RP-254b IV.B4.RP-256a IV.B4.RP-258 IV.B4.RP-259 Expansion Category IV.B4.RP-245b IV.B4.RP-246b IV.B4.RP-254b IV.B4.RP-260	N/A N/A IV.B4-4 (R-183) IV.B4-31 (R-205) N/A N/A N/A IV.B4-31 (R-205)
59	PWR (West.)	Stainless steel (SS, including CASS, PH SS or martensitic SS) or nickel alloy Westinghouse 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 dimension due to void swelling; 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 component-specific further evaluation recommendations for the components within this commodity group. However, generic further evaluation recommendations may apply. See Item 24 and SRP-LR Section 3.1.2.2.9.A.4 for evaluating inaccessible areas in partially accessible components.	~~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~~	~~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~~

Table 3.1-1 Summary of Aging Management Programsfor 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**
					See SRP-LR Section 3.1.2.2.9.A.5 if proposing reinspection frequencies for Westinghouse Expansion Category component , or for Westinghouse baffle-to-former bolts, that are in excess of the MRP recommended frequency of once every ten years. See SRP-LR Section 3.1.2.2.9.A.6 for evaluating the impact of thermal aging embrittlement on the fracture toughness property of the material if the IPA confirms the components are made from CASS, PH SS, or martensitic SS materials.	~~IV.B2.RP-386~~ ~~IV.B2.RO-388~~ No Additional Measures IV.B2.RP-267 Primary Category IV.B2.RP-270 IV.B2.RP-272 IV.B2.RP-276a IV.B2.RP-296 IV.B2.RP-297 IV.B2.RP-354 IV.B2.RP-388 Expansion Category 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 Existing Program IV.B2.RP-285 IV.B2.RP-288 IV.B2.RP-299	~~N/A~~ ~~N/A~~ N/A IV.B2-1 (R-124) IV.B2-6 (R-128) N/A N/A N/A N/A N/A 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 IV.B2-14 (R-137) IV.B2-18 (R-132) IV.B2-34 (R-115)

Table 3.1-1 Summary of Aging Management Programsfor 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**
108	PWR (B&W)	Stainless steel Babcock and Wilcox Core support assembly: upper flange welds exposed to reactor coolant and neutron flux	Cracking due to stress-corrosion cracking; loss of fracture toughness due to neutron irradiation embrittlement	Chapter XI.M16A, "PWR Vessel Internals"	Yes, basis for aging management is dependent on whether the IPA confirms a weld heat treatment /stress relief process was performed as part of the initial weld fabrication process. (See SRP-LR Section 3.1.2.2.9.D.1)	Primary Category / Preventative Monitoring IV.B4.RP-400 IV.B4.RP-401 (i.e., Primary Category if IPA confirms that fabrication of the welds were not subject to a heat treatment/stress relief process; otherwise PWR Vessel Internals Program to credit a "preventative/ mitigative monitoring program basis for these components if the IPA confirms prior implementation of the weld heat treatment)	 N/A N/A
109	PWR (B&W)	Babcock and Wilcox core barrel assemblies: internal baffle-to-baffle bolts, external baffle-to-baffle bolts and their locking devices, core	Cracking due to stress-corrosion cracking; loss of fracture toughness due to neutron irradiation	Chapter XI.M16A, "PWR Vessel Internals"	Yes, further evaluation of the basis for managing the relevant aging effects is recommended for these "Expansion	Expansion Category IV.B4.RP-243 IV.B4.RP-243a IV.B4.RP-244 IV.B4.RP-244a	 IV.B4-1 (R-128) N/A IV.B4-7 (R-125) N/A

Table 3.1-1 Summary of Aging Management Programsfor 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
		barrel-to-former bolts and their locking devices, former plates, and core barrel cylinder axial and circumferential welds exposed to reactor coolant and neutron flux	embrittlement; loss of preload due to irradiation-assisted stress relaxation or creep; and loss of materials due to wear		Category components. (See SRP-LR Section 3.1.2.2.9.D.2)	IV.B4.RP-250 IV.B4.RP-250a IV.B4.RP-375 IV.B4.RP-375a	IV.B4-12 (R-196) N/A N/A N/A

Appendix A, Section 4 – AMR Item Changes for Westinghouse-Design PWR RVI Components (GALL Section IV.B2 Changes)

B2. REACTOR VESSEL INTERNALS (PWR) - WESTINGHOUSE

Systems, Structures, and Components

This section addresses the Westinghouse pressurized-water reactor (PWR) vessel internals, ~~and~~ which consist~~s~~ of components in the upper internals assembly, the control rod guide tube assembly~~ies~~, the core barrel assembly, the baffle/former assembly, the lower internals assembly, lower support assembly, thermal shield assembly, and bottom mounted instrumentation system, and alignment and interfacing components. 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. AMR Items for Westinghouse PWR vessel internals are given in Table IV.B2.

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."

SRP-LR Further Evaluation "Acceptance Criteria" References for Westinghouse PWR Vessel Internals

The AMR items for Westinghouse PWR vessel internals (RVI components) in Table IV.B2 are based on and written to be consistent with the EPRI MRP's augmented inspection and evaluation (I&E) methodology in EPRI Report No. 1022863, "Materials Reliability Program: Pressurized Water Reactor Internals Inspection and Evaluation Guidelines (MRP-227-A), which was issued in December 2011 and approved in Revision 1 of the NRC safety evaluation on the methodology (Rev. 1 of the NRC SE on MRP-227; Refer to ADAMS ML11308A770). The EPR MRP methodology left some of the aging management bases for particular RVI components to be defined by the applicants or licensees implementing the methodology. The staff has identified these activities in either applicable Applicant/Licensee Action Items (A/LAIs) or Topical Report Condition Items (TRCIs) on the MRP's I&E methodology, as identified in the NRC SE on MRP-227, Revision 1, and has reflected these plant-specific activities in applicable further evaluation "acceptance criteria" subsections of NUREG-1800 (SRP-LR) Section 3.1.2.2. Therefore, the end of Table IV.B2 includes appropriate "Further Evaluation Recommendation

Notes for Westinghouse-design RVI Components” that identify the SRP-LR Section 3.1.2.2 further evaluation “acceptance criteria” recommendations that are applicable to the AMR items for Westinghouse PWR vessel internals. The “Further Evaluation” column entries in Table IV.B2 identify which of the further evaluation notes are applicable to the AMR items listed in the table.

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**	**Further Evaluation**
IV.B2.RP-300	IV.B2-33 (R-108)	Alignment and interfacing components: internals hold down spring	Stainless steel (Aust. SS material)	Reactor coolant and neutron flux	Loss of preload due to thermal and irradiation enhanced stress relaxation; loss of material due to wear	Chapter XI.M16A, "PWR Vessel Internals" ~~Primary components (identified in the "Structure and Components" column) no Expansion components~~ MRP Primary Category components without any Expansion Category links; MRP recommended physical measurement monitoring basis.	~~No~~ See Further Eval. Note 7.
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." ~~Existing Program components (identified in the "Structure and Components" column) no Expansion components~~ MRP Existing Program components without any Expansion Category component links. MRP invoking ASME Section XI VT-3 methods.	~~No~~ See Further Eval. Notes 3 and 6.

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**	**Further Evaluation**
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~~ MRP Existing Program components without any Expansion Category component links. MRP invoking ASME Section XI VT-3 methods.	~~No~~ See Further Eval. Note 3.
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 fatigue	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.B2.RP-273 and IV.B2.RP-286)~~ MRP Primary Category components using a MRP recommended UT volumetric technique. (See AMR Items IV.B2.RP-273 & IV.B2.RP-286	~~No~~ See Further Eval. Notes 3 and 4.

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**	**Further Evaluation**
						for the Expansion Category component links).	
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; changes in dimension due to void swelling; loss of preload due to thermal and irradiation enhanced stress relaxation	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)~~ MRP Primary Category components using a MRP recommended UT volumetric technique (See AMR Items IV.B2.RP-274 & IV.B2.RP-287 for the Expansion Category component links).	~~No~~ See Further Eval. Notes 3, 4, and 5.
IV.B2.RP-270	IV.B2-1(R-124)	Baffle-to-former assembly: baffle and former plates	Stainless steel	Reactor coolant and neutron flux	Change in dimension due to void swelling or distortion	Chapter XI.M16A, "PWR Vessel Internals" ~~Primary components (identified in the "Structure and Components" column) no Expansion components~~ MRP Primary Category components without any	~~No~~ See Further Eval. Note 3.

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**	**Further Evaluation**
						Expansion Category component links; MRP recommended VT-3 visual technique.	
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." MRP Primary Category components without any Expansion Category component links; MRP recommended VT-3 visual technique.	See Further Eval. Notes 3 and 6.
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 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~~ MRP Primary Category components without any Expansion Category component links; MRP recommended VT-3 visual technique.	~~No~~ See Further Eval. Notes 3 and 6.

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**	**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; change in dimension due to void swelling; loss of preload due to thermal and irradiation enhanced stress relaxation	Chapter XI.M16A, "PWR Vessel Internals." ~~Primary components (identified in the "Structure and Components" column) no Expansion components~~ MRP Primary Category components without any Expansion Category component links; MRP recommended VT-3 visual technique.	~~No~~ See Further Eval. Notes 3 and 5.
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 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-271)~~ MRP Expansion Category components using a MRP recommended UT volumetric technique (See AMR Item IV.B2.RP-271 for the Primary Category component link).	~~No~~ See Further Eval. Notes 3 and 4.

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**	**Further Evaluation**
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; change in dimension due to void swelling; loss of preload due to thermal and irradiation enhanced stress relaxation	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)~~ MRP Expansion Category components using a MRP recommended UT volumetric technique (See AMR Item IV.B2.RP-272 for the Primary Category component link).	~~No~~ See Further Eval. Notes 3, 4, and 5.
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." ~~Existing Program components (identified in the "Structure and Components" column) No expansion components; and Chapter XI.M37, "Flux Thimble Tube Inspection"~~ MRP Existing Program components invoking the plant's Existing	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**	**Further Evaluation**
						NRC Bulletin 88-09 Program/GALL AMP XI.M37 Flux Thimble Tube Inspection Program techniques (normally eddy current examinations).	
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 stress-corrosion cracking and fatigue	~~Chapter XI.M2, "Water Chemistry," and~~ Chapter XI.M16A, "PWR Vessel Internals" (and Chapter XI.M2, "Water Chemistry" if SCC is a plausible crack initiation mechanism) ~~Expansion components (identified in the "Structure and Components" column) (for Primary components see AMR Item IV.B2.RP-298)~~ MRP Expansion Category components using a MRP recommended EVT-1 visual techniques (See AMR Item IV.B2.RP-298 for the Primary Category component link).	~~No~~ See Further Eval. Notes 3 and 4.
IV.B2.RP-292		Bottom-mounted instrumentation system: bottom-mounted instrument~~ation~~ (BMI) column	Stainless steel	Reactor coolant and neutron flux	Loss of fracture toughness due to neutron irradiation embrittlement	~~Chapter XI.M2, "Water Chemistry," and~~ Chapter XI.M16A, "PWR Vessel Internals"	~~No~~ See Further Eval. Notes 3, 4, and 5.

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**	**Further Evaluation**
		bodies				~~Expansion components (identified in the "Structure and Components" column) (for Primary components see AMR Item IV.B2.RP-297)~~ <u>MRP Expansion Category components using a MRP recommended EVT-1 visual techniques (See AMR Item IV.B2.RP-297 for the Primary Category component link).</u>	
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)~~ <u>MRP Primary Category components without any expansion component links; MRP recommended VT-3 visual technique.</u>	~~No~~ <u>See Further Eval. Note 3.</u>
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 fatigue	Chapter XI.M2, “Water Chemistry,” and Chapter XI.M16A, “PWR Vessel Internals.”	~~No~~ <u>See Further Eval. Note 3.</u>

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**	**Further Evaluation**
						MRP Primary Category components using a MRP recommended EVT-1 visual technique ~~(identified in the "Structure and Components" column)~~ (~~for Expansion components~~ See AMR Items IV.B2.RP-291, IV.B2.RP-293, IV.B2.RP-291a, and IV.RP-291b for the Expansion Category component links).	
IV.B2.RP-297		Control rod guide tube (CRGT) assemblies: CRGT lower flange welds (accessible)	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." MRP Primary Category components using a MRP recommended EVT-1 visual technique~~(identified in the "Structure and Components" column)~~ (~~for Expansion components see~~ See AMR Items IV.B2.RP-290, ~~and~~ IV.B2.RP-292, IV.B2.RP-290a, and IV.B2.RP-290b for the Expansion Category component links).	~~No~~ See Further Eval. Notes 3 and 5.

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**	**Further Evaluation**
~~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		Control rod guide tube (CRGT) assemblies: guide tube support pins (split pins)	Nickel alloy or stainless steel	Reactor coolant and neutron flux	Cracking due to stress-corrosion cracking and fatigue	~~A plant-specific aging management program is to be evaluated~~ Chapter XI.M16A, "PWR Vessel Internals." Additionally, Chapter XI.M2, "Water Chemistry" for the SCC cracking mechanism. MRP Existing Program components with any Expansion Category component links; MRP invoking vendor-specific Existing Program techniques.	~~Yes, plant-specific~~ See Further Eval. Notes 1, 3, and 6.

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**	**Further Evaluation**
IV.B2.RP-356		Control rod guide tube (CRGT) assemblies: guide tube support pins (split pins)	Nickel alloy or stainless steel	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." MRP Existing Program components without any Expansion Category component links; MRP invoking vendor-specific techniques.	~~Yes, plant-specific~~ See Further Eval. Notes 1 and 3.
IV.B2.RP-387	N/A	Core barrel assembly: upper core barrel and lower core barrel circumferential (girth) welds ~~(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.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)~~ MRP Primary Category components using a MRP recommended EVT-1 visual technique (See AMR Item IV.B2.RP-387a for the Expansion Category component link).	~~No~~ See Further Eval. Note 3.

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**	**Further Evaluation**
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.M2, “Water Chemistry,” and Chapter XI.M16A, “PWR Vessel Internals.” MRP Expansion Category components using a MRP recommended EVT-1 visual technique (See AMR Item IV.B2.RP-387 for the Primary Category component link).	~~No~~ See Further Eval. Notes 3 and 4.
IV.B2.RP-388		Core barrel assembly: upper core barrel and lower core barrel circumferential (girth) welds ~~(axial 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)~~ MRP Primary Category components using a MRP recommended EVT-1 visual technique (See AMR Item IV.B2.RP-388a for the Expansion Category component link).	~~No~~ See Further Eval. Notes 3 and 5.

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	Further Evaluation
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." MRP Expansion Category components using a MRP recommended EVT-1 visual technique (See AMR Item IV.B2.RP-388 for the Primary Category component link).	See Further Eval. Notes 3, 4, and 5.
~~IV.B2.RP-282~~	~~IV.B2-8(R-120)~~	~~Core barrel assembly: core barrel flange~~	~~Stainless steel~~	~~Reactor coolant and neutron flux~~	~~Cracking due to stress corrosion cracking and 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-276)~~	~~No~~
IV.B2.RP-345		Core barrel assembly: core barrel flange (base metal)	Stainless steel	Reactor coolant and neutron flux	Cracking due to stress-corrosion cracking; 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~~ MRP Existing Program components without any Expansion Category	~~No~~ Further Eval. Notes 3 and 6.

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	Further Evaluation
						component links; MRP invokes the ASME Section XI VT-3 visual technique.	
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	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)~~ MRP Expansion Category components using a MRP recommended EVT-1 visual technique (See AMR Item IV.B2.RP-276 for the Primary Category component link).	~~No~~ See Further Eval. Notes 3 and 4.
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.M2, “Water Chemistry,” and Chapter XI.M16A, “PWR Vessel Internals.” MRP Expansion Category components using a MRP recommended EVT-1 visual technique (See AMR Item IV.B2.RP-276a for the Primary Category	See Further Eval. Notes 3, 4, and 5.

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**	**Further Evaluation**
						component link).	
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 and irradiation-assisted stress-corrosion cracking and fatigue	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)~~ MRP Primary Category components without any Expansion Category component links; MRP recommended EVT-1 visual technique.	~~No~~ See Further Eval. Note 3.
~~IV.B2.RP-281~~	~~IV.B2-9(R-122)~~	~~Core barrel assembly: lower core barrel flange weld~~	~~Stainless steel~~	~~Reactor coolant and neutron flux~~	~~Loss of fracture toughness due to 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**	**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.” ~~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)~~ MRP Primary Category components using a MRP recommended EVT-1 visual technique (See AMR Items IV.B2.RP-278 and IV.B2.RP-294 for the Expansion Category component links).	~~No~~ See Further Eval. Note 3.
IV.B2.RP-276a		Core barrel assembly: upper core barrel flange weld	Stainless steel	Reactor coolant and neutron flux	Loss of fracture toughness due to neutron irradiation embrittlement	Chapter XI.M2, “Water Chemistry,” and Chapter XI.M16A, “PWR Vessel Internals.” MRP Primary Category components using a MRP recommended EVT-1 visual	See Further Eval. Notes 3 and 5.

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**	**Further Evaluation**
						technique (See AMR Items IV.B2.RP-278a and IV.B2.RP-295 for the Expansion Category component links).	
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 fracture toughness due to neutron irradiation embrittlement	Chapter XI.M16A, “PWR Vessel Internals.” ~~Existing Program components (identified in the "Structure and Components" column) no Expansion components~~ MRP Existing Program components without any Expansion Category components; MRP invoking the ASME Section XI VT-3 visual technique.	~~No~~ See Further Eval. Notes 3 and 5.
IV.B2.RP-399		Lower internals assembly: clevis insert bolts or screws	Nickel alloy / stainless steel	Reactor coolant and neutron flux	Cracking due to irradiation assisted stress-corrosion cracking or fatigue	Chapter XI.M16A, “PWR Vessel Internals.” MRP Existing Program components without any Expansion Category components; MRP invoking the ASME Section XI VT-3 visual technique.	See Further Eval. Notes 3 and 6.

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**	**Further Evaluation**
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 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~~ MRP Existing Program components without any Expansion Category components; MRP invoking the ASME Section XI VT-3 visual technique.	~~No~~ See Further Eval. Notes 3 and 6.
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~~ MRP Existing Program components without any Expansion Category components; MRP invoking the ASME Section XI VT-3 visual technique.	~~No~~ See Further Eval. Notes 3 and 5.

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**	**Further Evaluation**
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.” ~~Expansion components (identified in the "Structure and Components" column) (for Primary components see AMR Item IV.B2.RP-298)~~ MRP Expansion Category components using a MRP recommended EVT-1 visual technique (See AMR Item IV.B2.RP-298 for the Primary Category component link).	~~No~~ See Further Eval. Notes 3 and 4.
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.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-297)~~ MRP Expansion Category components using a MRP recommended EVT-1 visual technique (See AMR Item	~~No~~ See Further Eval. Notes 3, 4, and 5.

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program	Further Evaluation
						IV.B2.RP-297 for the Primary Category component link).	
IV.B2.RP-291a		Lower support assembly: lower support forging or casting	Stainless steel (including CASS, PH SS, and mart. SS)	Reactor Coolant and Neutron Flux	Cracking due to stress-corrosion cracking and fatigue	Chapter XI.M16A, "PWR Vessel Internals." MRP Expansion Category components using a MRP recommended EVT-1 visual technique (See AMR Item IV.B2.RP-298 for the Primary Category component link)	See Further Eval. Notes 3 and 4.
IV.B2.RP-290a		Lower support assembly: lower support forging or casting	Stainless steel (including CASS, PH SS, and mart. SS)	Reactor Coolant and Neutron Flux	Loss of fracture toughness due to neutron irradiation embrittlement (and thermal aging embrittlement for CASS, PH SS, and mart. SS)	Chapter XI.M16A, "PWR Vessel Internals." MRP Expansion Category components using a MRP recommended EVT-1 visual technique (See AMR Item IV.B2.RP-297 for the Primary Category component link).	See Further Eval. Notes 3, 4, and 5.
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." ~~Expansion components (identified in the "Structure and Components" column)~~	~~No~~ See Further Eval. Notes 3 and 4.

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**	**Further Evaluation**
						~~(for Primary components see AMR Item IV.B2.RP-276)~~ MRP Expansion Category components using MRP recommended EVT-1 visual technique (See AMR Item IV.B2.RP-276 for the Primary Category component link).	
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" ~~Expansion components (identified in the "Structure and Components" column)~~ ~~(for Primary components see AMR Item IV.B2.RP-276)~~ MRP Expansion Category components using MRP recommended EVT-1 visual technique (See AMR Item IV.B2.RP-276a for the Primary Category component link).	~~No~~ See Further Eval. Notes 3, 4, and 5.
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 fatigue	Chapter XI.M2, "Water Chemistry," and Chapter XI.M16A, "PWR Vessel Internals." ~~Expansion components (identified in the~~	~~No~~ See Further Eval. Notes 3 and 4.

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**	**Further Evaluation**
						~~"Structure and Components" column)~~ ~~(for Primary components see AMR Item IV.B2.RP-271)~~ <u>MRP Expansion Category components using a MRP recommended UT volumetric technique (See AMR Item IV.B2.RP-271 for the Primary Category component link).</u>	
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	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)~~ <u>MRP Expansion Category components using a MRP recommended UT volumetric technique (See AMR Item IV.B2.RP-272 for the Primary Category component link).</u>	~~No~~ <u>See Further Eval. Notes 3, 4, and 5</u>.
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	Yes – TLAA <u>(See Further Eval Note 2)</u>

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	Further Evaluation
						the SRP, Section 4.3 “Metal Fatigue,” for acceptable methods for meeting the requirements of 10 CFR 54.21(c)(1).	
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		Reactor vessel internal components (inaccessible locations)	Stainless steel; nickel alloy	Reactor coolant and neutron flux	Cracking due to stress-corrosion cracking~~, and~~ ;irradiation-assisted stress-corrosion cracking; or fatigue	Chapter IX.M16A, “PWR Vessel Internals”; and additionally Chapter XI.M2, “Water Chemistry,” for SCC and IASSC mechanisms~~and Chapter XI.M16A, “PWR Vessel Internals~~”	Yes, if accessible Primary, Expansion or Existing program components indicate aging effects that need management (Refer to Further Eval. Note 3)
IV.B2.RP-269		Reactor vessel internal components (inaccessible locations)	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 IX.M16A, “PWR Vessel Internals”	Yes, if accessible Primary, Expansion or Existing program components indicate aging effects that need management (Refer to Further Eval. Note 3)

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**	**Further Evaluation**
IV.B2.RP-265		Reactor vessel internal components with no additional measures	Stainless steel; nickel alloy	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” 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		Reactor vessel internal components with no additional measures	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	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**	**Further Evaluation**
						Evaluation Guidelines"	
IV.B2.RP-382	IV.B2-26 (R-142)	Reactor vessel internals: core support structure components	Stainless steel; nickel alloy~~; cast austenitic stainless steel~~	Reactor coolant and neutron flux	Cracking due to fatigue; loss of material due to wear	Chapter XI.M1, "ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD"	No
IV.B2.RP-302		Thermal shield assembly: thermal shield flexures	Stainless steel	Reactor coolant and neutron flux	Cracking due to fatigue; loss of material due to wear	Chapter XI.M16A, "PWR Vessel Internals." ~~Primary components (identified in the "Structure and Components" column) no Expansion components~~ MRP Primary Category components without any Expansion Category component links; MRP recommended VT-3 visual technique.	See Further Eval. Notes 3 and 6.
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 and 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~~	See Further Eval. Notes 3 and 6.

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**	**Further Evaluation**
						MRP Existing Program components without any Expansion Category component links; MRP invoking the ASME Section XI VT-3 visual technique.	
IV,B2,RP-291b	N/A	Upper Internals Assembly; upper core plate	Stainless steel	Reactor coolant and neutron flux	Cracking due to stress-corrosion cracking or fatigue	Chapter IX.M16A, “PWR Vessel Internals”; and Chapter XI.M2, “Water Chemistry” MRP Expansion Category components using a MRP recommended EVT-1 visual technique (See AMR Item IV.B2.RP-298 for the Primary Category component link).	See Further Eval. Notes 3 and 4.
IV,B2,RP-290b	N/A	Upper Internals Assembly; upper 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 IX.M16A, “PWR Vessel Internals” MRP Expansion Category components using a MRP recommended EVT-1 visual technique (See AMR Item IV.B2.RP-297 for the Primary Category component link).	See Further Eval. Notes 3, 4, and 5.

Further Evaluation Recommendation Notes for Westinghouse-design RVI Components:

1. For AMR Items IV.B2.RP-355 and IV.B2.RP-356: Further evaluation of control rod guide tube (CRGT) assembly guide tube support pins (split pins) is recommended under Applicant/License Action Item (A/LAI) No. 3 in Revision 1 of the staff's safety evaluation (SE) of Topical Report MRP-227-A. Refer to SRP-LR Section 3.1.2.2.9.B, Part 1 (3.1.2.2.9.B.1) for the applicable SRP-LR further evaluation "acceptance criteria" recommendations.

2. For RVI components addressed in AMR Item IV.B2.RP-303: Yes - Further evaluation of RVI components with CUF or I_t fatigue analyses is recommended in accordance with A/LAI No. 8, Item 5 in the staff's SE on Topical Report MRP-227-A. The staff recommends further evaluation to determine whether the analyses conform to the definition of a TLAA in 10 CFR 54.3. Refer to the further evaluation recommendations in SRP-LR Section 3.1.2.2.1, as subject to the clarification in SRP-LR Section 3.1.2.2.9.A, Part 9 (3.1.2.2.9.A.9) on when CUF or I_t analyses for RVI components need to be identified as TLAAs for the LRA and when the PWR Vessel Internals Program may be used to accept a CUF or I_t fatigue TLAA for a given PWR RVI component in accordance with 10 CFR 54.21(c)(1)(iii).

3. For AMR IV.B2.RP-268 and IV.B2.RP-269, and other AMR Items in Table IV.B2 referring back to these AMR items: Yes – Further evaluation of inaccessible areas in partially accessible RVI components is recommended. Refer to SRP-LR Section 3.1.2.2.9.A, Part 4 (3.1.2.2.9.A.4) for the applicable SRP-LR further evaluation "acceptance criteria" recommendations.

4. For Westinghouse "Primary Category" baffle-to-former bolt components in AMR Items IV.B2.RP-270 and IV.B2.RP-271, and for Westinghouse "Expansion Category" components in AMR Items IV.B2.RP-273, IV.B2.RP-274, IVB2.RP-278, IV.B2.RP-278a, IV.B2.RP-286, IV.B2.RP-287, IV.B2.RP-290, IV.B2.RP-290a, IV.B2.RP-290b, IV.B2.RP-291, IV.B2.RP-291a, IV.B2.RP-291b, IV.B2.RP-292, IV.B2.RP-293, IV.B2.RP-294, IV.B2.RP-295, IV.B2.RP-387a, and IV.B2.RP-388a: Further evaluation is necessary if proposing reinspection frequency bases that exceed the MRP's recommended 10-year reinspection frequency basis for the components. Refer to SRP-LR Section 3.1.2.2.9.A, Part 5 (3.1.2.2.9.A.5) for the applicable SRP-LR further evaluation "acceptance criteria" recommendations.

5. For AMR items on loss of fracture toughness, further evaluation of the thermal aging embrittlement mechanism is necessary if the AMR item specifically calls out a CASS material, or for AMR items with general stainless steel references, if the applicant's IPA confirms that the stainless component is made from CASS, martensitic SS, or PH SS type materials. Refer to SRP-LR Section 3.1.2.2.9.A, Part 6 (3.1.2.2.9.A.6) for the applicable SRP-LR further evaluation "acceptance criteria" recommendations.

6. For AMR items in Table IV.B2 that credit VT-3 visual inspection methods to manage cracking in the components, further evaluation is necessary to justify the basis for crediting the VT-3 methods for the "detection of cracking. Refer to SRP-LR Section 3.1.2.2.9.A, Part 7 (SRP-LR 3.1.2.2.9.A.7) for the applicable SRP-LR further evaluation "acceptance criteria" recommendations.

7. For Westinghouse-design hold down springs in AMR item IV.B2.RP.300, further evaluation is recommended for the management of loss of preload/loss of compressibility in Westinghouse-design hold down springs. in accordance with A/LAI No. 5 in the staff's SE on Topical Report No. MRP-227-A. Refer to SRP-LR Section 3.1.2.2.9.B, Part 2 (3.1.2.2.9.B.2) for the applicable SRP-LR further evaluation "acceptance criteria" recommendations.

Appendix A, Section 5 – AMR Item Changes for CE-Design PWR RVI Components (GALL Section IV.B3 Changes)

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~~which consist~~s~~ of components in the upper internals assembly, the control element ~~assembly~~ (CEA) assembly~~shrouds~~, the core support barrel assembly, the core shroud assembly, and the lower support structure assembly, and incore instrumentation (ICI) components. 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. AMR Items for CE PWR vessel internals are given in Table IV.B3.

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."

SRP-LR Further Evaluation "Acceptance Criteria" References for CE PWR Vessel Internals

The AMR items for CE PWR vessel internals (RVI components) in Table IV.B3 are based on and written to be consistent with the EPRI MRP's augmented inspection and evaluation (I&E) methodology in EPRI Report No. 1022863, "Materials Reliability Program: Pressurized Water Reactor Internals Inspection and Evaluation Guidelines (MRP-227-A), which was issued in December 2011 and approved in Revision 1 of the NRC safety evaluation on the methodology (Rev. 1 of the NRC SE on MRP-227; Refer to ADAMS ML11308A770). The EPR MRP methodology left some of the aging management bases for particular RVI components to be defined by the applicants or licensees implementing the methodology. The staff has identified these activities in either applicable Applicant/Licensee Action Items (A/LAIs) or Topical Report Condition Items (TRCIs) on the MRP's I&E methodology, as identified in the NRC SE on MRP-227, Revision 1, and has reflected these plant-specific activities in applicable further evaluation "acceptance criteria" subsections of NUREG-1800 (SRP-LR) Section 3.1.2.2. Therefore, the end of Table IV.B3 includes appropriate "Further Evaluation Recommendation

Notes for CE-design RVI Components" that identify the SRP-LR Section 3.1.2.2 further evaluation "acceptance criteria" recommendations that are applicable to the AMR items for CE PWR vessel internals. The "Further Evaluation" column entries in Table IV.B3 identify which of the further evaluation notes are applicable to the AMR items listed in the table.

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	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 fatigue	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-313)~~ MRP Primary Category components using a MRP recommended VT-3 visual technique (See AMR Item IV.B3.RP-313 for the Expansion Category link).	~~No~~ See Further Eval. Notes 1, 3, and 6.
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 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.B3.RP-312)~~ MRP Expansion Category components using a MRP recommended VT-3 visual technique (See AMR Item IV.B3.RP-312 for the Primary Category link).	~~No~~ See Further Eval. Notes 1, 3, 4, and 6.

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**	**Further Evaluation**
IV.B3.RP-320	IV.B3-9 (R-162)	Core shroud assemblies (all plants): guide lugs; ~~and~~ guide lug 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~~ MRP Existing Program components without any Expansion Category component links; MRP invoking the existing ASME Section XI VT-3 visual techniques.	~~No~~ See Further Eval. Notes 3 and 6.
IV.B3.RP-319	IV.B3-9 (R-162)	Core shroud assemblies (all plants): guide lugs; ~~and~~ 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	Chapter XI.M16A, "PWR Vessel Internals." ~~Existing Program components (identified in the "Structure and Components" column) no Expansion components~~ MRP Existing Program components without any Expansion Category component links; MRP invoking the existing ASME Section XI VT-3 visual techniques.	~~No~~ See Further Eval. Note 3.
IV.B3.RP-358		Core shroud assemblies (for bolted core shroud assemblies): ~~(a) 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." ~~Expansion components~~	~~No~~ See Further Eval. Notes 3 and 6.

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**	**Further Evaluation**
		assembly components, including shroud plates and former plates				~~(identified in the "Structure and Components" column)~~ ~~(for Primary component see AMR Item IV.B3.RP-314)~~ MRP Primary Category components without any Expansion Category component links; MRP recommended VT-3 visual technique.	
IV.B3.RP-318	IV.B3-8 (R-163)	Core shroud assemblies (for bolted core shroud assemblies): ~~(a) shroud plates and (b) former plates~~ assembly components, including shroud plates and former plates	Stainless steel	Reactor coolant and neutron flux	Loss of fracture toughness due to neutron irradiation embrittlement; change in dimension due to void swelling	Chapter XI.M16A, "PWR Vessel Internals." ~~Primary components (identified in the "Structure and Components" column)~~ ~~no Expansion components~~ MRP Primary Category components without any Expansion Category component links; MRP recommended VT-3 visual technique.	~~No~~ See Further Eval. Notes 3 and 5.
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 and fatigue	Chapter XI.M2, "Water Chemistry," and Chapter XI.M16A, "PWR Vessel Internals." ~~Expansion components (identified in the "Structure and Components" column)~~ ~~(for Primary~~	~~No~~ See Further Eval. Notes 3 and 4.

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**	**Further Evaluation**
						~~components see AMR Item IV.B3.RP-314)~~ MRP Expansion Category components using a MRP recommended UT volumetric technique (See AMR Item IV.B3.RP-314 for the Primary Category component link).	
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; 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)~~ MRP Expansion Category components using a MRP recommended UT volumetric technique (See AMR Item IV.B3.RP-315 for the Primary Category component link).	~~No~~ See Further Eval. Notes 3, 4, and 5.
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 fatigue	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-316, IV.B3.RP-330, and~~	~~No~~ See Further Eval. Notes 3 and 4.

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**	**Further Evaluation**
						~~IV.B3.RP-358)~~ MRP Primary Category components using a MPR recommended UT volumetric technique (See AMR Items IV.B3.RP-316 and IV.B2.RP-330 for the Expansion Category component links).	
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; loss of fracture toughness due to neutron irradiation embrittlement; change in dimension due to void swelling	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)~~ MRP Primary Category components using a MPR recommended UT volumetric technique (See AMR Items IV.B3.RP-317 and IV.B2.RP-331 for the Expansion Category component links).	~~No~~ See Further Eval. Notes 3, 4, and 5.
IV.B3.RP-359		~~Core shroud assemblies (welded): (shroud plates and (b) former plates~~ Core shroud assembly (for CE-core shroud designs that are fabricated from two welded vertical sections): core shroud plate-to-former plate	Stainless steel	Reactor coolant and neutron flux	Loss of fracture toughness due to neutron irradiation embrittlement; change in dimension due to void swelling	Chapter XI.M16A, "PWR Vessel Internals." ~~Primary components (identified in the "Structure and Components" column) no Expansion components~~ MRP Primary Category components using a MRP recommended EVT-1 visual technique (See AMR	~~No~~ See Further Eval. Notes 3 and 5.

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**	**Further Evaluation**
		welds				Item IV.B3.RP-359a for the Expansion Category component link).	
IV.B3.RP-322		~~Core shroud assembly (for welded core shrouds in vertical sections): core shroud plate-former plate welds (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 shroud plate-to-former plate weld~~ Core shroud assembly (for CE-core shroud designs that are fabricated from two welded 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.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)~~ MRP Primary Category components using a MRP recommended EVT-1 visual technique (See AMR Item IV.B3.RP-323 for the Expansion Category component link).	~~No~~ See Further Eval. Note 3.
IV.B3.RP-326		Core shroud assembly (for welded CE core shroud designs ~~in~~that are assembled in two vertical sections): ~~gap between the upper and~~	Stainless steel	Reactor coolant and neutron flux	Change in dimension 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~~ See Further Eval. Notes 3, 5, and 7.

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	Further Evaluation
		~~lower plates~~ assembly components, including monitoring of the gap opening at the core shroud re-entrant corners				MRP Primary Category components without any Expansion Category component links; MRP recommended VT-1 visual technique (couple to physical measurement methods if a gap exists).	
IV.B3.RP-326a		Core shroud assembly (for welded CE core shroud designs that are assembled from 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." MRP Primary Category components without any Expansion Category component links; MRP recommended VT-1 visual technique (couple to physical measurement methods if a gap exists).	See Further Eval. Note 3.
IV.B3.RP-323		Core shroud assembly (for welded CE core shroud designs ~~in~~ that are assembled in two vertical sections): remaining axial welds ~~in shroud plate-to-former plate~~	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.” ~~Expansion components (identified in the "Structure and Components" column) (for Primary components see AMR Item IV.B3.RP-322)~~ MRP Expansion Category components using a MRP recommended EVT-1 visual technique (See AMR Items IV.B2.RP-322 and IV.B3.RP-324	~~No~~ See Further Eval. Notes 3 and 4.

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	Further Evaluation
						for the Primary Category component links).	
IV.B3.RP-359a		Core shroud assembly (for welded CE core shroud designs that are assembled in two vertical sections): remaining axial welds	Stainless steel	Reactor coolant and neutron flux	Loss of fracture toughness due to neutron irradiation embrittlement; change in dimension due to void swelling	Chapter XI.M2, “Water Chemistry,” and Chapter XI.M16A, “PWR Vessel Internals.” MRP Expansion Category components using a MRP recommended EVT-1 visual technique (See AMR Items IV.B2.RP-359 and IV.B3.RP-360 for the Primary Category component links).	See Further Eval. Notes 3, 4, and 5.
IV.B3.RP-324		Core shroud assembly (for welded CE core shroud designs assembled with full-height shroud plates): shroud plate 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~~	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)~~ MPR Primary Category components using a MRP recommended EVT-1 visual technique (See AMR Items IV.B3.RP-323 and IV.B3.RP-325 for the Expansion Category component links).	~~No~~ See Further Eval. Note 3.
IV.B3.RP-360		Core shroud assembly (for welded CE core shroud designs assembled	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~~	~~No~~ See Further Eval. Notes 3 and 5.

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program	Further Evaluation
		with full-height shroud plates): shroud plate 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~~				~~Components" column) (for Expansion components see AMR Item IV.B3.RP-361)~~ MPR Primary Category components using a MRP recommended EVT-1 visual technique (See AMR Items IV.B3.RP-359a and IV.B3.RP-361 for the Expansion Category component links).	
IV.B3.RP-325		Core shroud assembly (for welded CE core shroud designs assembled with full-height shroud plates): ~~remaining axial welds,~~ ribs~~,~~ and rings	Stainless steel	Reactor coolant and neutron flux	Cracking due to irradiation-assisted 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.B3.RP-324)~~ MPR Expansion Category components using a MRP recommended EVT-1 visual technique (See AMR Item IV.B3.RP-324 for the Primary Category component link).	~~No~~ See Further Eval. Notes 3 and 4.
IV.B3.RP-361		Core shroud assembly (for welded CE core shroud designs assembled with full-height shroud plates): ~~remaining axial welds,~~ ribs~~,~~	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~~	~~No~~ See Further Eval. Notes 3, 4, and 5.

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**	**Further Evaluation**
		and rings				~~components see AMR Item IV.B3.RP-360)~~ MPR Expansion Category components using a MRP recommended EVT-1 visual technique (See AMR Item IV.B3.RP-360 for the Primary Category component link).	
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." ~~Expansion components (identified in the "Structure and Components" column)~~ ~~(for Primary components see AMR Item IV.B3RP-327)~~ MRP Primary Category components using a MRP recommended EVT-1 visual technique (See AMR Item IV.B3.RP-362b for the Expansion Category component link).	~~No~~ See Further Eval. Notes 3 and 5.
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." MRP Primary Category components using a MRP recommended EVT-1 visual technique (See AMR Item IV.B3.RP-362c for the Expansion Category component link).	See Further Eval. Note 3.

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**	**Further Evaluation**
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." MRP Expansion Category components using a MRP recommended EVT-1 visual technique (See AMR Item IV.B3.RP-362 for the Primary Category component link).	See Further Eval. Notes 3, 4, and 5.
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." MRP Expansion Category components using a MRP recommended EVT-1 visual technique (See AMR Item IV.B3.RP-362a for the Primary Category component link).	See Further Eval. Notes 3 and 4.
IV.B3.RP-329	IV.B3-15(R-155)	Core support barrel assembly: ~~lower cylinder welds and remaining core barrel assembly welds~~ 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.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)~~ MRP Expansion Category components using a MRP recommended EVT-1 visual technique (See AMR Item IV.B3.RP-327 for the Primary Category Component link).	~~No~~ See Further Eval. Notes 3 and 4.

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	Further Evaluation
IV.B3.RP-333 (Apply this AMR only if: (1) the CLB does not include the design basis fatigue TLAA; or (2) the IPA confirms that either SCC, PWSCC, IASSC, or IGA is an additional plausible mechanism for inducing cracking in the component; or (3) the program's EVT-1 inspections will be used as the basis for accepting the TLAA in accordance with 10 CFR 54.21(c)(1(iii). – See Further Eval. Note 8)		Core support barrel assembly: lower flange weld~~, if fatigue life cannot be demonstrated by TLAA~~	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~~ MRP Primary Category components without any Expansion Category component links; MPR recommended EVT-1 visual technique (Refer to AMR Item IV.B3.RP-389 if the CLB includes a design basis fatigue TLAA	Yes~~, evaluate to determine the potential locations and extent of fatigue cracking~~ – See Further Eval. Note 8. See Further Eval. Note 3 if MRP-defined inspections will be performed.
IV.B3.RP-389 (Apply this AMR only if it can be demonstrated that cumulative fatigue damage can be adequately managed by a fatigue-based TLAA. However, also apply this AMRI item in conjunction with AMR Item		Core support barrel assembly: lower flange weld (if fatigue analysis exists)	Stainless steel	Reactor coolant and neutron flux	Cumulative fatigue damage due to fatigue; cracking 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 – See Further Eval. Note 2.

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**	**Further Evaluation**
IV.B3.RP-338 if if the EVT-1 examinations in the PWR Vessel Internals Program will be used to accept the TLAA in accordance with 10 CFR 54.21(c)(i)(iii))							
~~IV.B3.RP-328~~	~~IV.B3-15(R-155)~~	~~Core support barrel assembly: surfaces of the lower core barrel flange weld (accessible surfaces)~~	~~Stainless steel~~	~~Reactor coolant and neutron flux~~	~~Cracking due to stress corrosion cracking and 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~~	~~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." ~~Existing Program components (identified in the "Structure and Components" column) no Expansion components~~ MRP Existing Program components without any Expansion Category component links; MPR invoking existing ASME Code Section XI VT-3 visual examination requirements.	~~No~~ See Further Eval. Note 3.
IV.B3.RP-327	IV.B3-15(R-155)	Core support barrel assembly: upper core support barrel	Stainless steel	Reactor coolant and neutron flux	Cracking due to stress-corrosion cracking	Chapter XI.M2, "Water Chemistry," and Chapter XI.M16A, "PWR Vessel	~~No~~ See Further Eval. Note 3.

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**	**Further Evaluation**
		flange weld (accessible surfaces)				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)~~ MRP Primary Category components using a MRP recommended EVT-1 visual technique (See AMR Items IV.B3.RP-327a and IV.B3.RP-329 for the Expansion Category links).	
IV.B3.RP-357		Incore instruments (ICI): ICI thimble tubes ~~- lower~~	Zircaloy-4	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" MRP Existing Program components with no Expansion links; MRP referencing vendor or supplier condition monitoring bases for these components.	~~Yes, plant-specific~~ See Further Eval. Note 10.
IV.B3.RP-336	IV.B3-22(R-170)	Lower support structure~~: A286 fuel alignment pins~~ (all CE plants with core shroud 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	Chapter XI.M16A, "PWR Vessel Internals." ~~Existing Program components (identified in the "Structure and Components" column)~~ ~~no Expansion components~~ MRP Existing	~~No~~See Further Eval. Notes 3 and 5.

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**	**Further Evaluation**
					relaxation	Program components without any Expansion Category component links; MRP referencing the existing ASME Section XI VT-3 visual technique.	
IV.B3.RP-334	IV.B3-23(R-167)	Lower support structure~~: A286 fuel alignment pins~~ (all CE plants with core shroud designs assembled in two vertical sections): fuel alignment pins	Stainless steel	Reactor coolant and neutron flux	Cracking due to stress-corrosion cracking, irradiation-assisted stress-corrosion cracking, and fatigue	Chapter XI.M16A, “PWR Vessel Internals.” ~~Existing Program components (identified in the "Structure and Components" column)~~ ~~no Expansion components~~ MRP Existing Program components without any Expansion Category component links; MRP referencing the existing ASME Section XI VT-3 visual technique.	~~No~~ See Further Eval. Notes 3, 6, and 9.
IV.B3.RP-364		Lower support structure (applicable to all CE plant designs except those assembled with full-height shroud plates): core support column welds	~~Cast austenitic~~ Stainless steel	Reactor coolant and neutron flux	Loss of fracture toughness due to neutron irradiation embrittlement and thermal embrittlement	Chapter XI.M16A, "PWR Vessel Internals." ~~Expansion components (identified in the "Structure and Components" column)~~ ~~(for Primary components see AMR Item IV.B3RP-327)~~ MRP Primary Category components without any Expansion Category component links; MRP recommended VT-3 visual technique.	~~No~~ See Further Eval. Notes 3 and 5.

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**	**Further Evaluation**
IV.B3.RP-363		Lower support structure (applicable to all CE plant designs except those assembled with full-height shroud plates): core support column welds	Stainless steel	Reactor coolant and neutron flux	~~Loss of fracture toughness due to neutron irradiation embrittlement~~ Cracking due to stress-corrosion cracking, irradiation-assisted stress-corrosion cracking, and fatigue	Chapter XI.M16A, "PWR Vessel Internals." ~~Expansion components (identified in the "Structure and Components" column) (for Primary components see AMR Item IV.B3RP-327)~~ MRP Primary Category components without any Expansion Category links; MRP recommended VT-3 visual technique.	~~No~~ See Further Eval. Notes 3 and 6.
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 and 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.B3.RP-314)~~ MRP Expansion Category components using a MRP recommended UT volumetric technique (See AMR Item IV.B3.RP-314 for the Primary Category component link).	~~No~~ See Further Eval. Notes 3 and 4.

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**	**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." ~~Expansion components (identified in the "Structure and Components" column) (for Primary components see AMR Item 'IV.B3.RP-315)~~ MRP Expansion Category components using a MRP recommended UT volumetric technique (See AMR Item IV.B3.RP-315 for the Primary Category component link).	~~No~~ See Further Eval. Noted 3, 4, and 5.
~~IV.B3.RP-335~~	~~IV.B3-23(R-167)~~	~~Lower support structure: core support column welds, applicable to all plants except those assembled with full-height shroud plates~~	~~Stainless steel~~	~~Reactor coolant and neutron flux~~	~~Cracking due to stress corrosion cracking, irradiation-assisted stress corrosion cracking, and 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.B3.RP-327)~~	~~No.~~
IV.B3.RP-365		Lower support structure: 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~~ MRP Primary Category components without any Expansion	~~No~~ See Futher Eval. Notes 3 and 5.

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**	**Further Evaluation**
						Category component links; MPR recommended EVT-1 visual technique	
IV.B3.RP-343 (Apply this AMR only if: (1) the CLB does not include the design basis fatigue TLAA; or (2) the IPA confirms that either SCC, PWSCC, IASSC, or IGA is an additional plausible mechanism for inducing cracking in the component; or (3) the program's EVT-1 inspections will be used as the basis for accepting the TLAA in accordance with 10 CFR 54.21(c)(1(iii) – See Further Eval. Note 8.)		Lower support structure (applicable to CE plants designed with a core support plate): core support plate ~~(applicable to plants with a core support plate), if fatigue life cannot be demonstrated by TLAA~~	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~~ MRP Primary Category components without any Expansion Category component links; MPR recommended EVT-1 visual technique (Refer to AMR Item IV.B3.RP-390 if the CLB includes a design basis fatigue TLAA)	Yes~~, evaluate to determine the potential locations and extent of fatigue cracking~~ – See Further Eval. Note 8. See Further Eval. Note 3 if MRP-defined inspections will be performed.
IV.B3.RP-390 (Apply this AMR only if it can be demonstrated that cumulative fatigue damage can be adequately managed by a fatigue-based TLAA. However, also apply		Lower support structure (applicable to CE plants designed with a core support plate): core support plate ~~(applicable to plants with a core support plate), if fatigue analysis exists~~	Stainless steel	Reactor coolant and neutron flux	Cumulative fatigue damage due to fatigue; cracking by 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 – See Further Eval. Note 2.

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**	**Further Evaluation**
this AMRI item in conjunction with AMR Item IV.B3.RP-338 if if the EVT-1 examinations in the PWR Vessel Internals Program will be used to accept the TLAA in accordance with 10 CFR 54.21(c)(i)(iii))							
IV,B3.RP-327a		Lower support structure: lower support beams (other than deep beams in CE plants that are designed with shrouds fabricated from full height shroud plates)	Stainless steel	Reactor coolant and neutron flux	Cracking due to stress-corrosion cracking and fatigue	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)	See Further Eval. Notes 3 and 4..
IV.B3.RP-342		Lower support structure: deep beams (applicable assemblies with full height shroud plates)	Stainless steel	Reactor coolant and neutron flux	Cracking due to stress-corrosion cracking, irradiation-assisted stress-corrosion cracking, and 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~~ MRP Primary Category components without any Expansion Category component links. MRP recommended EVT-1 visual technique.	~~No~~ See Futher Eval. Note 3.

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**	**Further Evaluation**
IV.B3.RP-366		Lower support structure (Applicable to all CE plants with core shrouds assembled with full height shroud plates): deep beams ~~(applicable assemblies with full height shroud 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)~~ ~~no Expansion components~~ MRP Primary Category components without any Expansion Category component links. MRP recommended EVT-1 visual technique.	~~No~~ See Further Eval. Notes 3 and 5.
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 – See Further Eval. Note 2.
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-309		Reactor vessel internal components (inaccessible locations)	Stainless steel; nickel alloy	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"	Yes, if accessible Primary, Expansion or Existing program components indicate aging effects that need management (Refer to Further Eval. Note 3)

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**	**Further Evaluation**
IV.B3.RP-311		Reactor vessel internal components (inaccessible locations)	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 (Refer to Further Eval. Note 3)
IV.B3.RP-306		Reactor vessel internal components with no additional measures	Stainless steel; nickel alloy	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” 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.RP-307		Reactor vessel internal components with no additional measures	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	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	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**	**Further Evaluation**
					enhanced stress relaxation; loss of material due to wear	Reliability Program: Pressurized Water Reactor Internals Inspection and Evaluation Guidelines"	
IV.B3.RP-382	IV.B3-22(R-170)	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.B3.RP-338 (Apply this AMR only if: (1) the CLB does not include the design basis fatigue TLAA; or (2) the IPA confirms that either SCC, PWSCC, IASSC, or IGA is an additional plausible mechanism for inducing cracking in the component; or (3) the program's EVT-1 inspections will be used as the basis for accepting the TLAA in accordance with 10 CFR 54.21(c)(1(iii) – See Further Eval. Note 8.)		Upper internals assembly (applicable to all CE plants with core shrouds assembled with full height shroud plates): fuel alignment plate ~~(applicable to plants with core shrouds assembled with full height shroud plates), if fatigue life cannot be demonstrated by TLAA~~	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~~ MRP Primary Category components without any Expansion Category component links; MPR recommended EVT-1 visual technique (Refer to AMR Item IV.B3.RP-391 if the CLB includes a design basis fatigue TLAA)	Yes~~, evaluate to determine the potential locations and extent of fatigue cracking~~ – See Further Eval. Note 8. See Further Eval. Note 3 if MRP-defined inspections will be performed.

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**	**Further Evaluation**
IV.B3.RP-391 (Apply this AMR only if it can be demonstrated that cumulative fatigue damage can be adequately managed by a fatigue-based TLAA. However, also apply this AMRI item in conjunction with AMR Item IV.B3.RP-338 if if the EVT-1 examinations in the PWR Vessel Internals Program will be used to accept the TLAA in accordance with 10 CFR 54.21(c)(i)(iii))		Upper internals assembly (applicable to CE plants with core shrouds assembled with full height shroud plates): fuel alignment plate ~~(applicable to plants with core shrouds assembled with full height shroud plates), if fatigue analysis exists~~	Stainless steel	Reactor coolant and neutron flux	Cumulative fatigue damage due to fatigue; cracking 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 – See Further Eval. Note 2.
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 additionally for cracking due to SCC or IASSC, Chapter XI.M2 “Water Chemistry.” MRP referring to plant-specific or vendor-specific condition monitoring bases.	See Further Eval. Note 10.

Further Evaluation Recommendation Notes for CE-design RVI Components:

1. For AMR Items IV.B3.RP-312 and IV.B3.RP-313: Further evaluation of control element assembly (CEA) instrument guide tubes is recommended. Refer to the staff’s further evaluation “acceptance criteria” recommendations in SRP-LR Section 3.1.2.2.9.C, Part 1 (3.1.2.2.9.C.1). Perform in conjunction with in conjunction with Further Evaluation Note No. 6 below and the NRC further evaluation “acceptance criteria” recommendations in SRP-LR Section 3.1.2.2.9.A, Part 7 (3.1.2.2.9.A.7).

2. For AMR Item IV.B3.RP-339, IV.B3.RP-389, IV.B3.RP-390, and IV.B3.RP-391: Yes - Further evaluation of RVI components with CUF or I_t fatigue analyses is recommended in accordance with A/LAI No. 8, Item 5 in the staff’s SE

on Topical Report MRP-227-A. The staff recommends further evaluation to determine whether the analyses conform to the definition of a TLAA in 10 CFR 54.3. Refer to the further evaluation recommendations in SRP-LR Subsection 3.1.2.2.1, as subject to the clarification in SRP-LR Subsection 3.1.2.2.9.A, Part 9 (3.1.2.2.9.A.9) on when CUF or I_t analyses for RVI components need to be identified as TLAAs for the LRA and when the PWR Vessel Internals Program may be used to accept a CUF or I_t fatigue TLAA for a given PWR RVI component in accordance with 10 CFR 54.21(c)(1)(iii).

3. For AMR Items IV.B3.RP-309 and IV.B3.RP-311, and other AMR Items in Table IV.B3 referring back to these AMR items: Further evaluation of inaccessible areas in partially accessible RVI components is recommended. Refer to the further evaluation recommendations in SRP-LR Subsection 3.1.2.2.9.A, Part 4 (3.1.2.2.9.A.4) for the applicable SRP-LR further evaluation “acceptance criteria” recommendations.

4. For Combustion Engineering-design (CE-design) “Primary Category” core shroud bolt components in bolted design CE core shroud assemblies in AMR Items IV.B3.RP-314 and IV.B3.RP-315 and for CE-design “Expansion Category” components in AMR Items IV.B3.RP-313, IV.B3.RP-316, IV.B3.RP-317, IV.B3.RP-323, IV.B3.RP-325, IV.B3.RP-327a, IV.B3.RP-329, IV.B3.RP-330, IV.B3.RP-331, IV.B3.RP-359a, IV.B3.RP-361, IV.B3.RP-362b, and IV.B3.RP-362c: Further evaluation is necessary if proposing reinspection frequency bases that exceed the MRP’s recommended 10-year reinspection frequency basis for the components. Refer to the further evaluation recommendations in SRP-LR Section 3.1.2.2.9.A, Part 5 (3.1.2.2.9.A.5) for the applicable SRP-LR further evaluation “acceptance criteria” recommendations.

5. For the AMR items in Table IV.B3 on loss of fracture toughness, further evaluation of the thermal aging embrittlement mechanism is necessary if the AMR item specifically calls out a CASS material, or for AMR items with general stainless steel references, if the applicant’s IPA confirms that the stainless component is made from CASS, martensitic SS, or PH SS type materials. Refer to the further evaluation recommendations in SRP-LR Section 3.1.2.2.9.A, Part 6 (3.1.2.2.9.A.6) for the applicable SRP-LR further evaluation “acceptance criteria” recommendations.

6. For AMR items in Table IV.B3 that credit VT-3 visual inspection methods to manage cracking in the components, further evaluation is necessary to justify the basis for crediting the VT-3 methods for the “detection of cracking. Refer to SRP-LR Section 3.1.2.2.9.A, Part 7 (SRP-LR 3.1.2.2.9.A.7) for the applicable SRP-LR further evaluation “acceptance criteria” recommendations.

7. For AMR item IV.B3.RP-326: Further evaluation is recommended to manage changes in dimension or distortion in the gaps area of CE core shrouds that are assembled in two vertical segments. Refer to SRP-LR Section 3.1.2.2.9.C, Part 1 (3.1.2.2.9.C.1) for the applicable SRP-LR further evaluation “acceptance criteria” recommendations.

8. For AMR Items IV.B3.RP-333, IV.B3.RP-338, and IV.B3.RP-343: Further evaluation of the core flange welds in the core support barrel assembly, the core support plate in the lower core support assembly (applicable to CE-designed reactors that include a core support plate), and the fuel alignment plates is recommended in order to determine whether the components have been analyzed with either a CUF or I_t fatigue analysis under Further Evaluation Note No. 2 above, and is so, whether the TLAA can demonstrate that fatigue-induced cracking will not initiate in the components during the period of extended operation. Otherwise apply the AMR items if the IPA determines that the TLAAs cannot demonstrate adequate acceptance or management of fatigue-induced cracking or cumulative fatigue damage in the components, or if SCC-induced cracking, IASCC-induced cracking, or intergranular attack are plausible aging effect mechanisms for the components. Refer to SRP-LR Section 3.1.2.2.9.C, Part 3 (3.1.2.2.9.C.3) for the applicable SRP-LR further evaluation “acceptance criteria” recommendations.

9. For AMR Items IV.B3.RP-334 and IV.B3.RP-336: Further evaluation the fuel alignment pins in CE-designed facilities is recommended. Refer to SRP-LR Section 3.1.2.2.9.C, Part 4 (3.1.2.2.9.C.4) for the applicable SRP-LR further evaluation “acceptance criteria” recommendations, and coordinate in conjunction with Further Evaluation Note No. 6 above and with the further evaluation “acceptance criteria” recommendations in SRP-LR Section 3.1.2.2.9.A, Part 7 (3.1.2.2.9.A.7) on when VT-3 visual examination methods may be used to monitor for cracking in a component.

10. For AMR Items IV.B3.RP-357 and IV.B3.RP-400: Further evaluation of CE-design incore instrument (ICI) flux thimble tubes and thermal shield positioning pins is recommended. Refer to SRP-LR Section 3.1.2.2.9.C, Part 5 (3.1.2.2.9.C.5) for the applicable SRP-LR further evaluation “acceptance criteria” recommendations.

Appendix A, Section 6 – AMR Item Changes for B&W-Design PWR RVI Components (GALL Section IV.B2 Changes)

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~~ which consist~~s~~ of components in the plenum cover assembly~~and plenum cylinder~~, 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. 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. AMR Items for B&W PWR vessel internals are given in Table IV.B4.

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."

SRP-LR Further Evaluation "Acceptance Criteria" References for B&W PWR Vessel Internals

The AMR items for B&W PWR vessel internals (RVI components) in Table IV.B4 are based on and written to be consistent with the EPRI MRP's augmented inspection and evaluation (I&E) methodology in EPRI Report No. 1022863, "Materials Reliability Program: Pressurized Water Reactor Internals Inspection and Evaluation Guidelines (MRP-227-A), which was issued in December 2011 and approved in Revision 1 of the NRC safety evaluation on the methodology (Rev. 1 of the NRC SE on MRP-227; Refer to ADAMS ML11308A770). The EPR MRP methodology left some of the aging management bases for particular RVI components to be defined by the applicants or licensees implementing the methodology. The staff has identified these activities in either applicable Applicant/Licensee Action Items (A/LAIs) or Topical Report Condition Items (TRCIs) on the MRP's I&E methodology, as identified in the NRC SE on MRP-227, Revision 1, and has reflected these plant-specific activities in applicable further evaluation "acceptance criteria" subsections of NUREG-1800 (SRP-LR) Section 3.1.2.2.

Therefore, the end of Table IV.B4 includes appropriate "Further Evaluation Recommendation Notes for B&W-design RVI Components" that identify the SRP-LR Section 3.1.2.2 further evaluation "acceptance criteria" recommendations that are applicable to the AMR items for B&W PWR vessel internals. The "Further Evaluation" column entries in Table IV.B4 identify which of the further evaluation notes are applicable to the AMR items listed in the table.

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

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program	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 (CASS)	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)~~ MPR Primary Category components without any Expansion Category component links; MRP recommended VT-3 visual examination technique.	~~No~~ See Further Eval. Notes 2 and 4.
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, irradiation-assisted stress-corrosion cracking or fatigue	Chapter XI.M16A, "PWR Vessel Internals." MPR Primary Category components without any Expansion Category component links; MRP recommended VT-3 visual examination technique.	See Further Eval. Notes 2 and 5.

IV REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM B4 Reactor Vessel Internals (PWR) – Babcock and Wilcox							
Item	**Link**	**Structure and/or Component**	**Material**	**Environment**	**Aging Effect/ Mechanism**	**Aging Management Program**	**Further Evaluation**
IV.B4.RP-245	IV.B4-13 (R-194)	Core barrel assembly (applicable to CR-3 or DB only): ~~(a) upper thermal shield bolts; (b) surveillance specimen holder tube bolts (Davis-Besse, only); (c) surveillance specimen tube holder studs, and nuts (Crystal River Unit 3, only)~~ surveillance specimen holder tube (SSHT) studs/nuts or 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" ~~Expansion components (identified in the "Structure and Components" column) (for Primary components see AMR Items IV.B4.RP-247 and IV.B4.RP-248)~~ MRP Expansion Category components using a MRP recommended UT volumetric technique (See AMR Items IV.B4.RP-247, IV.B4.RP-248, and IV.B4.RP-256 for the Primary Category component links).	~~No~~ See Further Eval. Notes 2 and 3.

IV REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM B4 Reactor Vessel Internals (PWR) – Babcock and Wilcox							
Item	**Link**	**Structure and/or Component**	**Material**	**Environment**	**Aging Effect/ Mechanism**	**Aging Management Program**	**Further Evaluation**
IV.B4.RP-245a		Core barrel assembly (applicable to CR-3 or DB only): surveillance specimen holder tube (SSHT) stud or bolt locking devices	Stainless steel; nickel alloy	Reactor coolant and neutron flux	Cracking due to fatigue	Chapter XI.M16A, "PWR Vessel Internals." MRP Expansion Category components using a MRP recommended VT-3 visual technique (See AMR Items IV.B4.RP-247a, IV.B4.RP-248a, and IV.B4.RP-256a for the Primary Category component links).	See. Further Eval. Notes 2, 3, and 5.
IV.B4.RP-245b		Core barrel assembly (applicable to CR-3 or DB only): surveillance specimen holder tube (SSHT) stud or bolt locking devices	Stainless steel; nickel alloy	Reactor coolant and neutron flux	Loss of material due to wear; Changes in dimension due to void swelling or distortion	Chapter XI.M16A, "PWR Vessel Internals." MRP Expansion Category components using a MRP recommended VT-3 visual technique (See AMR Items IV.B4.RP-247b, IV.B4.RP-248b, and IV.B4.RP-256b for the Primary Category component links).	See Further Eval. Notes 2 and 3.

IV REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM B4 Reactor Vessel Internals (PWR) – Babcock and Wilcox							
Item	**Link**	**Structure and/or Component**	**Material**	**Environment**	**Aging Effect/ Mechanism**	**Aging Management Program**	**Further Evaluation**
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.” ~~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)~~ MRP Primary Category components using a MRP recommended UT volumetric technique (See AMR Items IV.B4.RP-245, IV.B4.RP-246, and IV.B4.RP-254 for the Expansion Category component links).	~~No~~ See Further Eval. Note 2.

IV REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM B4 Reactor Vessel Internals (PWR) – Babcock and Wilcox							
Item	**Link**	**Structure and/or Component**	**Material**	**Environment**	**Aging Effect/ Mechanism**	**Aging Management Program**	**Further Evaluation**
IV.B4.RP-247a		Core barrel assembly: accessible 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." MRP Primary Category components using a MRP recommended VT-3 visual technique (See AMR Items IV.B4.RP-245a, IV.B4.RP-246a, and IV.B4.RP-254a for the Expansion Category component links).	See Further Eval. Notes 2 and 5.
IV.B4.RP-247b		Core barrel assembly: accessible lower core barrel (LCB) bolt locking devices	Stainless steel; nickel alloy	Reactor coolant and neutron flux	Loss of material due to wear; Changes in dimension due to void swelling or distortion	Chapter XI.M16A, "PWR Vessel Internals." MRP Primary Category components using a MRP recommended VT-3 visual technique (See AMR Items IV.B4.RP-245b, IV.B4.RP-246b, and IV.B4.RP-254b for the Expansion Category component links).	See Further Eval. Note 2.

IV REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM B4 Reactor Vessel Internals (PWR) – Babcock and Wilcox							
Item	**Link**	**Structure and/or Component**	**Material**	**Environment**	**Aging Effect/ Mechanism**	**Aging Management Program**	**Further Evaluation**
IV.B4.RP-249	IV.B4-12 (R-196)	Core barrel assembly: baffle plates ~~accessible surfaces within one inch around each baffle plate flow and bolt hole~~	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)~~ MPR Primary Category components using a MRP recommended VT-3 visual technique (See AMR Item IV.B4.RP-250 Expansion Category component link).	~~No~~ See Further Eval. Notes 2 and 4.
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" MPR Primary Category components using a MRP recommended VT-3 visual technique (See AMR Item IV.B4.RP-250a Expansion Category component link).	See Further Eval. Notes 2 and 5.

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program	Further Evaluation
IV.B4.RP-241	IV.B4-7 (R-125)	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	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) (for Expansion components see AMR Items IV.B4.RP-244 and IV.B4.RP-375)~~ MRP Primary Category components using a MRP recommended UT volumetric technique (See AMR Items IV.B4.RP-244 and IV.B4.RP-375 for the Expansion Category component links).	~~No~~ See Further Eval. Notes 2 and 3.
IV.B4.RP-241a		Core barrel assembly: 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.M16A, “PWR Vessel Internals.” MRP Primary Category components using a MRP recommended VT-3 visual technique (See AMR Item IV.B4.RP-244a for the Expansion Category component link).	See Further Eval. Notes 2, 3, and 5.

IV REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM B4 Reactor Vessel Internals (PWR) – Babcock and Wilcox							
Item	**Link**	**Structure and/or Component**	**Material**	**Environment**	**Aging Effect/ Mechanism**	**Aging Management Program**	**Further Evaluation**
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; 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.)~~ MRP Primary Category components using a MRP recommended UT volumetric technique (See AMR Items IV.B4.RP-243 and IV.B4.RP-375a for the Expansion Category component links).	~~No~~See Further Eval. Notes 2, 3, and 4.
IV.B4.RP-240a		Core barrel assembly: accessible 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 preload due to thermal and irradiation enhanced stress relaxation; loss of material due to wear	Chapter XI.M16A, "PWR Vessel Internals." MRP Primary Category components using a MRP recommended VT-3 visual technique (See AMR Item IV.B4.RP-243a for the Expansion Category component link).	See Further Eval. Notes 2, 3, and 4.

IV REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM B4 Reactor Vessel Internals (PWR) – Babcock and Wilcox							
Item	**Link**	**Structure and/or Component**	**Material**	**Environment**	**Aging Effect/ Mechanism**	**Aging Management Program**	**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)~~ MRP Expansion Category components using supplement evaluation or replacement activities as the alternative to VT-3 visual inspections under the expanded scope (See IV.B4.RP-249 for the Primary Category component link).	~~No~~See Further Eval. Notes 4 and 8.

IV REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM B4 Reactor Vessel Internals (PWR) – Babcock and Wilcox							
Item	**Link**	**Structure and/or Component**	**Material**	**Environment**	**Aging Effect/ Mechanism**	**Aging Management Program**	**Further Evaluation**
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." MRP Expansion Category components using supplement evaluation or replacement activities as the alternative to VT-3 visual inspections under the expanded scope (See IV.B4.RP-249a for the Primary Category component link).	See Further Eval. Note 8.

IV REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM B4 Reactor Vessel Internals (PWR) – Babcock and Wilcox							
Item	**Link**	**Structure and/or Component**	**Material**	**Environment**	**Aging Effect/ Mechanism**	**Aging Management Program**	**Further Evaluation**
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 or 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.B4.RP-241)~~ MRP Expansion Category components using supplement evaluation or replacement activities as the alternative to UT volumetric inspections under the expanded scope (See IV.B4.RP-241 for the Primary Category component link).	~~No~~See Further Eval. Note 7.

IV REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM B4 Reactor Vessel Internals (PWR) – Babcock and Wilcox							
Item	**Link**	**Structure and/or Component**	**Material**	**Environment**	**Aging Effect/ Mechanism**	**Aging Management Program**	**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; loss of material due to wear	Chapter XI.M16A, "PWR Vessel Internals." MRP Expansion Category components using supplement evaluation or replacement activities as the alternative to UT volumetric inspections under the expanded scope (See IV.B4.RP-240 for the Primary Category component link).	~~No~~See Further Eval. Notes 4 and 7.

IV REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM B4 Reactor Vessel Internals (PWR) – Babcock and Wilcox							
Item	**Link**	**Structure and/or Component**	**Material**	**Environment**	**Aging Effect/ Mechanism**	**Aging Management Program**	**Further Evaluation**
IV.B4.RP-244	IV.B4-7 (R-125)	Core barrel assembly; ~~(a)~~ external baffle-to-baffle bolts~~;~~ ~~(b)~~ and 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, fatigue, and overload	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-241)~~ MRP Expansion Category components using supplement evaluation or replacement activities as the alternative to UT volumetric inspections under the expanded scope (See IV.B4.RP-241 for the Primary Category component link).	~~No~~See Further Eval. Note 8.

IV REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM B4 Reactor Vessel Internals (PWR) – Babcock and Wilcox							
Item	**Link**	**Structure and/or Component**	**Material**	**Environment**	**Aging Effect/ Mechanism**	**Aging Management Program**	**Further Evaluation**
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, fatigue, and overload	Chapter XI.M2, "Water Chemistry," and Chapter XI.M16A, "PWR Vessel Internals." MRP Expansion Category components using supplement evaluation or replacement activities as the alternative to VT-3 visual inspections under the expanded scope (See IV.B4.RP-241a for the Primary Category component link).	See Further Eval. Note 8.

IV REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM B4 Reactor Vessel Internals (PWR) – Babcock and Wilcox							
Item	**Link**	**Structure and/or Component**	**Material**	**Environment**	**Aging Effect/ Mechanism**	**Aging Management Program**	**Further Evaluation**
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~~ 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; 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)~~ MRP Expansion Category components using supplement evaluation or replacement activities as the alternative to UT volumetric inspections under the expanded scope (See IV.B4.RP-240 for the Primary Category component link).	~~No~~See Further Eval. Note 8.

IV REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM B4 Reactor Vessel Internals (PWR) – Babcock and Wilcox							
Item	**Link**	**Structure and/or Component**	**Material**	**Environment**	**Aging Effect/ Mechanism**	**Aging Management Program**	**Further Evaluation**
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 preload due to thermal and irradiation enhanced stress relaxation; loss of material due to wear	Chapter XI.M16A, "PWR Vessel Internals." MRP Expansion Category components using supplement evaluation or replacement activities as the alternative to VT-3 visual inspections under the expanded scope (See IV.B4.RP-240a for the Primary Category component link).	Yes – See Further Eval. Note 8.

IV REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM B4 Reactor Vessel Internals (PWR) – Babcock and Wilcox							
Item	**Link**	**Structure and/or Component**	**Material**	**Environment**	**Aging Effect/ Mechanism**	**Aging Management Program**	**Further Evaluation**
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.” ~~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)~~ MRP Primary Category components using a MRP recommended UT volumetric technique (See AMR Items IV.B4.RP-245, IV.B4.RP-246, and IV.B4.RP-254 for the Expansion Category component links).	~~No~~ See Further Eval. Note 2.

IV REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM B4 Reactor Vessel Internals (PWR) – Babcock and Wilcox							
Item	**Link**	**Structure and/or Component**	**Material**	**Environment**	**Aging Effect/ Mechanism**	**Aging Management Program**	**Further Evaluation**
IV.B4.RP-248a		Core support shield (CSS) assembly: accessible 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.” MRP Primary Category components using a MRP recommended VT-3 visual technique (See AMR Items IV.B4.RP-245a, IV.B4.RP-246a, and IV.B4.RP-254a for the Expansion Category component links).	See Further Eval. Notes 2 and 5.
IV.B4.RP-248b		Core support shield (CSS) assembly: accessible upper core barrel (UCB) bolt locking devices	Stainless steel; nickel alloy	Reactor coolant and neutron flux	Loss of material due to wear; Changes in dimension due to void swelling or distortion	Chapter XI.M16A, “PWR Vessel Internals.” MRP Primary Category components using a MRP recommended VT-3 visual technique (See AMR Items IV.B4.RP-245b, IV.B4.RP-246b, and IV.B4.RP-254b for the Expansion Category component links).	See Further Eval. Note 2.

IV REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM B4 Reactor Vessel Internals (PWR) – Babcock and Wilcox							
Item	**Link**	**Structure and/or Component**	**Material**	**Environment**	**Aging Effect/ Mechanism**	**Aging Management Program**	**Further Evaluation**
~~IV.B4.RP-253~~	~~IV.B4-21 (R-191)~~	~~Core support shield (CSS) assembly: (a) CSS cast outlet nozzles (Oconee Unit 3 and Davis-Besse, only); (b) CSS vent valve discs~~	~~Cast austenitic stainless steel~~	~~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	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 bottom retaining ring~~ CSS vent valve top and bottom retaining rings (valve body components)	Stainless steel	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) No Expansion components~~ MRP Primary Category components without any Expansion Category component links. MRP recommended VT-3 visual technique.	~~No~~ See Further Eval. Notes 2 and 4.

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

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program	Further Evaluation
IV.B4.RP-252a	IV.B4-16 (R-188)	Core support shield (CSS) assembly: CSS vent valve top and bottom retaining rings (valve body components)	Stainless steel	Reactor coolant and neutron flux	Cracking due to SCC or fatigue	Chapter XI.M2, “Water Chemistry,” and Chapter XI.M16A, “PWR Vessel Internals” MRP Primary Category components without any Expansion Category component links. MRP recommended VT-3 visual technique.	See Further Eval. Notes 2 and 5.
IV.B4.RP-251	IV.B4-15 (R-190)	Core support shield (CSS) assembly: CSS top flange~~; 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 (core clamping)	Chapter XI.M16A, “PWR Vessel Internals” ~~Primary component (identified in the "Structure and Components" column) No Expansion components~~ MRP Primary Category components without any Expansion Category links; MRP recommended one-time physical measurement at least two refueling outages prior to the PEO, followed by periodic VT-3 visual examinations.	~~No~~ See Further Eval. Note 2.

IV REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM B4 Reactor Vessel Internals (PWR) – Babcock and Wilcox							
Item	**Link**	**Structure and/or Component**	**Material**	**Environment**	**Aging Effect/ Mechanism**	**Aging Management Program**	**Further Evaluation**
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 (core clamping)	Chapter XI.M16A, "PWR Vessel Internals." MRP Primary Category components without any Expansion Category links; MRP recommended one-time physical measurement at least two refueling outages prior to the PEO, followed by periodic VT-3 visual examinations.	See Further Eval. Note 2.

IV REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM B4 Reactor Vessel Internals (PWR) – Babcock and Wilcox							
Item	**Link**	**Structure and/or Component**	**Material**	**Environment**	**Aging Effect/ Mechanism**	**Aging Management Program**	**Further Evaluation**
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) (for Primary components see AMR Items IV.B4.RP-247 and IV.B4.RP-248)~~ MRP Primary Category components using a MRP recommended UT volumetric technique (See AMR Items IV.B4.RP-245, IV.B4.RP-246, and IV.B4.RP-254 for the Expansion Category component links).	~~No~~ See Further Eval. Note 2.

IV REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM B4 Reactor Vessel Internals (PWR) – Babcock and Wilcox							
Item	**Link**	**Structure and/or Component**	**Material**	**Environment**	**Aging Effect/ Mechanism**	**Aging Management Program**	**Further Evaluation**
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." MRP Primary Category components using a MRP recommended VT-3 visual technique (See AMR Items IV.B4.RP-245a, IV.B4.RP-246a, and IV.B4.rp-254a for the Expansion Category component links).	See Further Eval. Notes 2 and 5.
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 dimension due to distortion or void swelling	Chapter XI.M16A, "PWR Vessel Internals." MRP Primary Category components using a MRP recommended VT-3 visual technique (See AMR Items IV.B4.RP-245b, IV.B4.RP-246b, and IV.B4.rp-254b for the Expansion Category component links).	See Further Eval. Note 2.

IV REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM B4 Reactor Vessel Internals (PWR) – Babcock and Wilcox							
Item	**Link**	**Structure and/or Component**	**Material**	**Environment**	**Aging Effect/ Mechanism**	**Aging Management Program**	**Further Evaluation**
IV.B4.RP-259	IV.B4-31 (R-205)	Incore Monitoring Instrumentation (IMI) guide tube assembly: ~~accessible top surfaces of~~ IMI guide tube spider-to-lower grid rib section~~s~~ welds	Stainless steel; nickel alloy	Reactor coolant and neutron flux	Loss of fracture toughness due to ~~thermal aging,~~ neutron irradiation embrittlement	Chapter XI.M16A, “PWR Vessel Internals” ~~Primary components (identified in the "Structure and Components" column) (for Expansion components see Item IV.B4.RP-260)~~ Primary Category components using a MRP recommended VT-3 visual technique (See AMR Item IV.B4.RP-260 for the Expansion Category component link).	~~No~~ See Further Eval. Notes 2 and 4.
IV.B4.RP-259a		Incore Monitoring Instrumentation (IMI) guide tube assembly: IMI guide tube spider-to-lower grid rib sections welds	Stainless steel; nickel alloy	Reactor coolant and neutron flux	Cracking due to stress-corrosion cracking	Chapter XI.M16A, “PWR Vessel Internals” Primary Category components using a MRP recommended VT-3 visual technique (See AMR Item IV.B4.RP-260a for the Expansion Category component link).	See Further Eval. Notes 2 and 5.

IV REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM B4 Reactor Vessel Internals (PWR) – Babcock and Wilcox							
Item	**Link**	**Structure and/or Component**	**Material**	**Environment**	**Aging Effect/ Mechanism**	**Aging Management Program**	**Further Evaluation**
IV.B4.RP-258	IV.B4-4 (R-183)	Incore Monitoring Instrumentation (IMI) guide tube assembly: ~~accessible top surfaces of IMI Incore~~ IMI guide tube spiders (castings)	Cast austenitic stainless steel	Reactor coolant and neutron flux	Loss of fracture toughness due to thermal aging~~,~~ embrittlement 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)~~ Primary Category components using a MRP recommended VT-3 visual technique (See AMR Item IV.B4.RP-260 for the Expansion Category component link).	~~No~~ See Further Eval. Notes 2 and 4.
IV.B4.RP-258a		Incore Monitoring Instrumentation (IMI) guide tube assembly: IMI guide tube spiders	Stainless steel; nickel alloy	Reactor coolant and neutron flux	Cracking due to stress-corrosion cracking	Chapter XI.M16A, "PWR Vessel Internals" Primary Category components using a MRP recommended VT-3 visual technique (See AMR Item IV.B4.RP-260a for the Expansion Category component link).	See Further Eval. Notes 2 and 5.

IV REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM B4 Reactor Vessel Internals (PWR) – Babcock and Wilcox							
Item	**Link**	**Structure and/or Component**	**Material**	**Environment**	**Aging Effect/ Mechanism**	**Aging Management Program**	**Further Evaluation**
IV.B4.RP-254	IV.B4-25 (R-210)	Lower grid assembly: alloy X-750 lower grid shock pad bolts and locking devices (TMI-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) (for Primary components see AMR Items IV.B4.RP-247 and IV.B4.RP-248)~~ MRP Expansion Category components using a MRP recommended UT volumetric technique (See AMR Items IV.B4.RP-247, IV.B4.RP-248 and IV.B4.RP-256 for the Primary Category component links).	~~No~~ See Further Eval. Notes 2 and 3.

IV REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM B4 Reactor Vessel Internals (PWR) – Babcock and Wilcox							
Item	**Link**	**Structure and/or Component**	**Material**	**Environment**	**Aging Effect/ Mechanism**	**Aging Management Program**	**Further Evaluation**
IV.B4.RP-254a		Lower grid assembly: alloy X-750 lower grid shock pad bolt locking devices (TMI-1, only)	Nickel alloy	Reactor coolant and neutron flux	Cracking due to fatigue	Chapter XI.M16A, "PWR Vessel Internals," MRP Expansion Category components using a MRP recommended VT-3 visual technique (See AMR Items IV.B4.RP-247a, IV.B4.RP-248a and IV.B4.RP-256a for the Primary Category component links).	See Further Eval. Notes 2, 3, and 5.
IV.B4.RP-254b		Lower grid assembly: alloy X-750 lower grid shock pad bolts and locking devices (TMI-1, only)	Nickel Alloy	Reactor coolant and neutron flux	Loss of material due to wear; Changes in dimension due to void swelling or distortion	Chapter XI.M16A, "PWR Vessel Internals," MRP Expansion Category components using a MRP recommended VT-3 visual technique (See AMR Items IV.B4.RP-247b, IV.B4.RP-248b, and IV.B4.RP-256b for the Primary Category component links).	See Further Eval. Notes 2 and 3.

IV REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM B4 Reactor Vessel Internals (PWR) – Babcock and Wilcox							
Item	**Link**	**Structure and/or Component**	**Material**	**Environment**	**Aging Effect/ Mechanism**	**Aging Management Program**	**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.M2, “Water Chemistry,” and Chapter XI.M16A, “PWR Vessel Internals.” ~~Expansion components (identified in the "Structure and Components" column) (for Primary components see AMR Items IV.B4.RP-247 and IV.B4.RP-248)~~ MRP Expansion Category components using a MRP recommended UT volumetric technique (See AMR Items IV.B4.RP-247, IV.B4.RP-248, and IV.B4.RP-256 for the Primary Category component links).	~~No~~ See Further Eval. Notes 2 and 3.
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	MRP Expansion Category components using a MRP recommended VT-3 visual technique (See AMR Items IV.B4.RP-247a, IV.B4.RP-248a, and IV.B4.RP-256a for the Primary Category component links).	See Further Eval. Notes 2, 3, and 5.

IV REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM B4 Reactor Vessel Internals (PWR) – Babcock and Wilcox							
Item	**Link**	**Structure and/or Component**	**Material**	**Environment**	**Aging Effect/ Mechanism**	**Aging Management Program**	**Further Evaluation**
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 dimension due to void swelling or distortion	Chapter XI.M16A, "PWR Vessel Internals." MRP Expansion Category components using a MRP recommended VT-3 visual technique (See AMR Items IV.B4.RP-247b, IV.B4.RP-248b, and IV.B4.RP-256b for the Primary Category component links).	See Further Eval. Notes 2 and 3.
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)~~ Expansion Category components using a MRP recommended VT-3 visual technique (See AMR Items IV.B4.RP-258 and IV.B4.RP-259 for the Primary Category links).	~~No~~ See Further Eval. Notes 2, 3, and 4.

IV REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM B4 Reactor Vessel Internals (PWR) – Babcock and Wilcox							
Item	**Link**	**Structure and/or Component**	**Material**	**Environment**	**Aging Effect/ Mechanism**	**Aging Management Program**	**Further Evaluation**
IV.B4.RP-260a		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	Cracking due to stress-corrosion cracking or fatigue	Chapter XI.M16A, "PWR Vessel Internals" Expansion Category components using a MRP recommended VT-3 visual technique (See AMR Items IV.B4.RP-258a and IV.B4.RP-259a for the Primary Category links).	Further Eval. Notes 2, 3, and 5.

IV REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM B4 Reactor Vessel Internals (PWR) – Babcock and Wilcox							
Item	**Link**	**Structure and/or Component**	**Material**	**Environment**	**Aging Effect/ Mechanism**	**Aging Management Program**	**Further Evaluation**
IV.B4.RP-262	IV.B4-32 (R-203)	Lower grid assembly: accessible alloy X-750 dowel-to-lower fuel assembly support pad 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.” ~~Expansion components (identified in the "Structure and Components" column) (for Primary components see AMR Item IV.B4.RP-261)~~ MRP Expansion Category components using a MRP recommended VT-3 visual technique (See AMR Item IV,B4.RP-261 for the Primary Category component link).	~~No~~ See Further Eval. Notes 2, 3, and 5.

IV REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM B4 Reactor Vessel Internals (PWR) – Babcock and Wilcox							
Item	**Link**	**Structure and/or Component**	**Material**	**Environment**	**Aging Effect/ Mechanism**	**Aging Management Program**	**Further Evaluation**
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)~~ MRP Primary Category components using a MRP recommended VT-3 visual technique (See AMR Items IV,B4.RP-262 and IV.B4.RP-352 for the Expansion Category component links).	~~No~~ See Further Eval. Notes 2 and 5.
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– See Further Eval. Note 1.

IV REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM B4 Reactor Vessel Internals (PWR) – Babcock and Wilcox							
Item	**Link**	**Structure and/or Component**	**Material**	**Environment**	**Aging Effect/ Mechanism**	**Aging Management Program**	**Further Evaluation**
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		Reactor vessel internal components (inaccessible locations)	Stainless steel; nickel alloy	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"	Yes, if accessible Primary, Expansion or Existing program components indicate aging effects that need management (Refer to Further Eval. Note 2)

IV REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM B4 Reactor Vessel Internals (PWR) – Babcock and Wilcox							
Item	**Link**	**Structure and/or Component**	**Material**	**Environment**	**Aging Effect/ Mechanism**	**Aging Management Program**	**Further Evaluation**
IV.B4.RP-239		Reactor vessel internal components (inaccessible locations)	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 <u>(Refer to Further Eval. Note 2)</u>
IV.B4.RP-236		Reactor vessel internal components with no additional measures	Stainless steel; nickel alloy	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" 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 REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM B4 Reactor Vessel Internals (PWR) – Babcock and Wilcox							
Item	**Link**	**Structure and/or Component**	**Material**	**Environment**	**Aging Effect/ Mechanism**	**Aging Management Program**	**Further Evaluation**
IV.B4.RP-237		Reactor vessel internal components with no additional measures	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 REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM B4 Reactor Vessel Internals (PWR) – Babcock and Wilcox							
Item	**Link**	**Structure and/or Component**	**Material**	**Environment**	**Aging Effect/ Mechanism**	**Aging Management Program**	**Further Evaluation**
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)~~ MRP Expansion Category components using a MRP recommended VT-3 visual technique (See AMR Item IV.B4.RP-261 for the Primary Category component link).	~~No~~ Yes – Apply Further Eval. Note 5. Further evaluation is necessary to justify the use of VT-3 visual techniques for the detection of cracking in the components. IPA Dependent – See Further Eval. Note 2. Further evaluation is necessary if IPA confirms that components are only partially accessible to the inspection technique (Refer to AMR IV.B3.RP-238) IPA Dependent – See Further Eval Note 3. Further evaluation is necessary if proposing an inspection frequency in excess of once every 10 years.
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 IX.M2, “Water Chemistry.” Primary Category components using a MRP recommended EVT-1 visual technique if IPA confirms that weld was not subject to a post-weld stress relief/heat treatment process.	Yes– See Further Eval Note 6. IPA Dependent – See Further Eval. Note 2. Further evaluation is necessary if inspections will be performed and IPA confirms that components are only partially accessible to the inspection technique (Refer to AMR IV.B3.RP-238).

IV REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM B4 Reactor Vessel Internals (PWR) – Babcock and Wilcox							
Item	**Link**	**Structure and/or Component**	**Material**	**Environment**	**Aging Effect/ Mechanism**	**Aging Management Program**	**Further Evaluation**
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,” and IX.M2, “Water Chemistry.” Primary Category components using a MRP recommended EVT-1 visual technique if IPA confirms that weld was not subject to a post-weld stress relief/heat treatment process.	Yes– See Further Eval Note 6. IPA Dependent – See Further Eval. Note 2. Further evaluation is necessary if inspections will be performed and IPA confirms that components are only partially accessible to the inspection technique (Refer to AMR IV.B3.RP-238). IPA Dependent – See Further Eval. Note 4. Further evaluation of thermal aging embrittlement is necessary if inspections will be performed and IPA confirms that components are fabricated from either CASS, PH SS, or marternsitic SS.

Further Evaluation Recommendation Notes for B&W-design RVI Components:

1. For RVI components addressed in AMR Item IV.B4.R-53: Yes - Further evaluation of RVI components with CUF or I_t fatigue analyses is recommended in accordance with A/LAI No. 8, Item 5 in the staff’s SE on Topical Report MRP-227-A. The staff recommends further evaluation to determine whether the analyses conform to the definition of a TLAA in 10 CFR 54.3. Refer to the further evaluation recommendations in SRP-LR Section 3.1.2.2.1, as subject to the clarification in SRP-LR Section 3.1.2.2.9.A, Part 9 (3.1.2.2.9.A.9) on when CUF or I_t analyses for RVI components need to be identified as TLAAs for the LRA and when the PWR Vessel Internals Program may be used to accept a CUF or I_t fatigue TLAA for a given PWR RVI component in accordance with 10 CFR 54.21(c)(1)(iii).

2. For IV.B4.RP-238 and IV.B4.RP-239, and other AMR Items in Table IV.B2 referring back to these AMR items: Yes – Further evaluation of inaccessible areas in partially accessible RVI components is recommended. Refer to SRP-LR Section 3.1.2.2.9.A, Part 4 (3.1.2.2.9.A.4) for the applicable SRP-LR further evaluation “acceptance criteria” recommendations.

3. For Babcock and Wilcox-design (B&W-design) “Primary Category” baffle-to-former bolt components in AMR Items IV.B4.RP-240 and IV.B4.RP-241 and for B&W-design “Expansion Category” components in AMR Items IV.B4.RP-245, IV.B4.RP-245a, IV.B4.RP-245b, IV.B4.RP-246, IV.B4.RP-246a, IV.B4.RP-246b, IV,B4.RP-254, IV.B4.RP-254a, IV.B4.RP-254b, IV.B4.RP-260, IV.B4.RP-260a, IV.B4.RP-262, and IV.B4.RP-352: Further evaluation is necessary if proposing reinspection frequency bases that exceed the MRP’s recommended 10-year reinspection frequency basis for the components. Refer to SRP-LR Section 3.1.2.2.9.A, Part 5 (3.1.2.2.9.A.5) for the applicable SRP-LR further evaluation “acceptance criteria” recommendations.

4. For AMR items on loss of fracture toughness, further evaluation of the thermal aging embrittlement mechanism is necessary if the AMR item specifically calls out a CASS material, or for AMR items with general stainless steel references, if the applicant's IPA confirms that the stainless component is made from CASS, martensitic SS, or PH SS type materials. Refer to SRP-LR Section 3.1.2.2.9.A, Part 6 (3.1.2.2.9.A.6) for the applicable SRP-LR further evaluation "acceptance criteria" recommendations.

5. For AMR items in Table IV.B4 that credit VT-3 visual inspection methods to manage cracking in the components, further evaluation is necessary to justify the basis for crediting the VT-3 methods for the "detection of cracking. Refer to SRP-LR Section 3.1.2.2.9.A, Part 7 (SRP-LR 3.1.2.2.9.A.7) for the applicable SRP-LR further evaluation "acceptance criteria" recommendations.

6. For AMR Items IV.B4.RP-400 and IV.B4.RP-401: Further evaluation of B&W core support shield assembly upper flange welds is recommended for management of cracking in the components. Refer to SRP-LR Section 3.1.2.2.9.D, Part 1 (SRP-LR 3.1.2.2.9.D.1) for the applicable SRP-LR further evaluation "acceptance criteria" recommendations.

7. For AMR Items IV.B4.RP-375 and IV.B4.RP-375a: Further evaluation of internal baffle-to-baffle bolts (Expansion Category components) is necessary because in Report MRP-227-A, the MRP identified that the industry has yet to define inspection techniques for the internal baffle-to-baffle bolts in B&W-designed reactors, even thought the bolts are accessible for inspection. Refer to SRP-LR Section 3.1.2.2.9.D, Part 2 (SRP-LR 3.1.2.2.9.D.2) for the applicable SRP-LR further evaluation "acceptance criteria" recommendations.

8. For B&W Expansion Category components in AMR Items IV.B4.RP-243, IV.B4.RP-243a, IV.B4.RP-244, IV.B4.RP-244a, IV.B4.RP-250, and IV.B4.RP-250a: The AMR items relate to the management of age-related degradation mechanisms in the following components in B&W-design reactors: (1) external baffle-to-baffle bolts and their locking devices; (2) core barrel-to-former bolts and their locking devices; (3) former plates; and (4) circumferential welds [girth welds] and vertical welds [axial welds] in the core barrel. Further evaluation of these Expansion Category components is necessary because in Report MRP-227-A, the MRP identified that the components are inaccessible for inspection. Refer to SRP-LR Section 3.1.2.2.9.D, Part 2 (SRP-LR 3.1.2.2.9.D.2) for the applicable SRP-LR further evaluation "acceptance criteria" recommendations.

Appendix A - Section 7: Staff Proposed Revisions to Definitions in Table IX.C, Selected Definitions & Use of Terms for Describing and Standardizing, MATERIALS

Changes to the "Definition as used in this document" column entry for the "Stainless Steel" Material Definition in GALL Table IX.C

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 steel materials 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 generically applicable to all of the various stainless steel categories, it can be assumed that citing of "stainless steel" in the "Material" column of the AMR item emcompasses all of those stainless steel types. ~~the recommended AMP is the same for PH stainless steel or 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. The EPRI MRP has also reported in MRP-227-A 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 aging embrittlement is known to be 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 referenced in the "material" column of applicable AMR line item.~~Therefore, when this aging effect is being considered, CASS is specifically 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, and Types 304, 304NG, 308, 308L, 309, 309L, 316, 347, 403, and 416. Examples of CASS designations include CF-3, -8, -3M, and -8M. [Ref. 6, 7]~~Examples of wrought austenitic stainless materials include Type 304, 304NG, 304L, 308, 308L, 309, 309L, 316 and 347 stainless steels. Table 2.10 of the American Welding Society (AWS) *Welding Handbook*, Volume 4, Seventh Edition [Copyright AWS, 1982 - Library of Congress No. 75-28707] provides a more comprehensive list of wrought austenitic stainless steel material types. Examples of CASS include CF3, CF3M, CF8 and CF8M material types. Table 2.11 of the AWS *Welding Handbook* provides a more comprehensive list of CASS material types, In addition, the AWS Welding Handbook provides examples of the following additional materials: (a) Type 403, 410, 420, and 431 martensitic stainless steels; (b) Type 15-5, 17-4, and 13-8-Mo PH stainless steels, and (c) SA-193, Grade B8 and B8M bolting materials.

Appendix B

List of Acronyms Commonly Used In LR-ISG No. 2011-04

Acronym List

Acronym	*Definition*
A/LAI	applicant/licensee action item
AMP	aging management program
AMR	aging management review
ANO-1	reference to the Arkansas Nuclear One Station, Unit 1 reactor
ASME	American Society of Mechanical Engineers
BMI	bottom mounted instrumentation (a Westinghouse-design instrumentation reference)
B&W	Babcock and Wilcox (a nuclear NSSS vendor)
CASS	cast austenitic stainless steel
CE	Combustion Engineering (a nuclear NSSS vendor)
CEA	control element assembly
CRGT	control rod guide tube
CR-3	reference to the Crystal River Station, Unit 3 reactor
DB	reference to the Davis Besse Nuclear Power Plant
EPRI	Electric Power Research Institute
EVT-1	enhanced visual testing (an enhanced version of the VT-1 visual method)
FD	flow distributor (a B&W-design reference)
FSAR	final safety analysis report
GALL	Generic Aging Lessons Learned (Report – in reference to NUREG-1801, Rev.2)
IASCC	irradiation-assisted stress-corrosion cracking
IGA	intergranular attack
ICI	incore instrumentation (a CE-design instrumentation reference)
IMI	incore monitoring instrumentation (a B&W-design instrumentation reference)
IPA	integrated plant assessment
LCB	lower core barrel (a B&W-design reference)
LST	lower thermal shield (a B&W-design reference)
MRP	Materials Reliability Program
NEI	Nuclear Energy Institute
ONS-1	reference to the Oconee Nuclear Station, Unit 1 reactor
ONS-3	reference to the Oconee Nuclear Station, Unit 3 reactor
PH	precipitation hardened
PWR	pressurized-water reactor
RVI	reactor vessel internal
SRP-LR	Standard Review Plan for License Renewal Applications for Nuclear Power Plants (in reference to NUREG-1800, Revision 2)
SS	stainless steel
SCC	stress-corrosion cracking
SSHT	surveillance specimen holder tube
TLAA	time limited aging analysis
TMI/TMI-1	reference to the Three Mile Island Nuclear Station, Unit 1 reactor
TRCI	Topical Report Condition Item
TS	Technical Specifications
UCB	upper core barrel (a B&W-design reference)
UTS	upper thermal shield (a B&W-design reference)
UFSAR	updated final safety analysis report (an updated version of the FSAR)
UT	ultrasonic testing (a volumetric non-destructive examination method)

Acronym List

Acronym	*Definition*
VT-1	a type of visual examination technique that is defined in Subsection IWA of the ASME Boiler and Pressure Vessel Code, Section XI, Division 1
VT-3	a type of visual examination technique that is defined in Subsection IWA of the ASME Boiler and Pressure Vessel Code, Section XI, Division 1
XL	extra long (refers to a type of core plate in Westinghouse designed reactors)
X-750	reference to a specific type of nickel alloy material
10 CFR	Title 10, *Code of Federal Regulations*