ATTACHMENT 1

Responses to NRC Requests for Additional Information (RAIs) on WCAP-17096-NP, Rev. 3

BACKGROUND:

The NRC staff's requirements for including passive, long-lived PWR vessel internals within the scope of a reactor vessel internals aging management program (RVI AMP) are discussed in Section 54.21 of Title 10, *Code of Federal Regulation* (10 CFR 54.21). The NRC staff's current aging management criteria and guidance for these types of programs are presented in Chapter XI.M16A, "PWR Vessel Internals," in NUREG-2192, Volume 2 (Reference (Ref.) 1).

NOTES:

(1) The abbreviated and italicized term "TR" in this document refers to the Topical Report WCAP-17096, Revision 3 report (Ref. 2). The term "MRP" refers to Materials Reliability Program

(2) The bases for NRC staff's conditions associated with the prior approved version of this report, WCAP-17096-NP-A, Revision 2 (Ref. 3) were included in its Final Safety Evaluation (SE) (Ref. 4) for Revision 2 of the WCAP.

Request for Additional Information (RAI) 01

Applicable TR Sections, Appendices, Tables, Figures, or Pages:

TR Section 1 (TR Page 1-1)

Background and Issue:

In the first added paragraph on TR Page 1-1, the PWROG includes the following statement indicating that the TR methodology may be used in conjunction with the original inspection and evaluation (I&E) methodology in MRP-227-A (Ref. 5):

> "*Note that the methodologies defined herein can also be used in conjunction with Rev. 0 of MRP-227-A [1] after accounting for differences in component application, component nomenclature, and component numbering.*"

The staff determined that this may create an inconsistency with the EPRI MRP's "Needed Requirement," as cited consistent with "Needed requirement" criteria in the latest version of NEI 03-08, for converting licensees' PWR RVI AMPs over to the methodology in MRP-227, Rev. 1-A (Ref. 6). Specifically, Section 7.3 of MRP-227, Rev. 1-A establishes an NEI "Needed requirement" that states that plants shall convert their AMPs over to the I&E methodology in MRP-227, Rev. 1-A by "January 1, 2022." Therefore, licensees' PWR RVI AMPs may follow the original I&E methodology in MRP-227-A report only until December 31, 2021, after which, the AMPs would need to implement MRP-227, Rev. 1-A to address the "Needed requirement" in Section 7.3 of MRP-227 Rev. 1-A.

NRC Request:

Justify the purpose for including the above quoted statement on TR Page 1-1 given the "Needed requirement" implementation in Section 7.3 of MRP-227, Rev. 1-A by January 1, 2022.

Response to RAI 01:

The purpose of this TR is to identify the NRC-approved acceptance criteria and evaluation procedures to evaluate degradation identified during MRP-227 inspections. Section 7.5 of MRP-227, Rev. 1-A [6] identifies the applicable "Needed" NEI 03-08 requirement:

> *"Examination results that do not meet the examination acceptance criteria defined in Section 5 of these guidelines shall be recorded and entered in the owner's plant corrective action program and dispositioned. Engineering evaluations used to disposition an examination result that does not meet the examination acceptance criteria in Section 5, shall be conducted in accordance with NRC approved evaluation methods (i.e., ASME Code Section XI, PWR Owners Group topical report WCAP-17096-NP-A or equivalent method)."*

This "Needed" NEI 03-08 requirement does not include an implementation date for when a specific revision of WCAP-17096-NP must be used. Rather, the TR clarifies the changes and revisions to the acceptance criteria that must be used:

> *"Therefore, each utility must remain cognizant of changes made to the methodologies defined herein, and are advised to update the acceptance criteria when substantive changes occur, especially for changes that are mandated by NEI 03-08 [45]."*

Therefore, the latest NRC-approved version of WCAP-17096-NP-A should be used independent of the revision of MRP-227.

Section 7.3 of MRP-227, Rev. 1-A [6] identifies the implementation date for MRP-227, Rev. 1-A as "January 1, 2022." However, Section 7.3 of MRP-227, Rev. 1-A allows units that have submitted an AMP to the regulator under MRP-227-A, with a period of extended operation beginning before January 1, 2022, to continue to use MRP-227-A for the unit's baseline MRP-227 inspection. Therefore, the statement described in this RAI was included in the TR to provide guidance on the use of WCAP-17096-NP, Rev. 3 for plants that are allowed to continue to apply MRP-227-A.

Reference 36 of the TR will be revised to reflect the NRC-approval of MRP-227, Rev. 1-A as shown on page 5 of Attachment 2.

RAI 04

Applicable TR Sections, Appendices, Tables, Figures, or Pages:
TR Section 2.2[1] (TR Page 2-6); TR Appendix A, Component-specific assessments in B&W-ID Items A.1.9 and A.2.11; TR Appendix C, Component-specific assessments in CE-ID Items 2, 2.1, 3, 3.1, 5, 5.1, 5.2, 5.3, 5.4, 6, 6.1, 6.2, 7, and 11; and TR Appendix E, Component-specific assessments in W-ID Items 3, 3.1, 3.2, 3.3, 4, 4.2, and 4.3.

Background and Issue:
The TR cites the EPRI report identified in TR Reference No. 46 (Ref. 7 for this RAI set) for establishing crack growth rates (CGRs) of stainless steel components. The TR states that this EPRI report provides the basis for ASME Code Case N-889. Code Case N-889 has yet to be approved for use in the current version of NRC Regulatory Guide (RG) 1.147 (Ref. 13), as incorporated by reference in 10 CFR 50.55a; the staff's review of this Code Case is currently pending. Even if the staff does approve Code Case N-889, application of the Code Case may be subject to certain conditions in RG 1.147, which could affect the suitability for use and application of the referenced EPRI report under the criteria defined for the report in the TR.

NRC Request:
Justify the use of the EPRI report (TR Reference 46) as the methodology for establishing CGRs for irradiated stainless steel materials, given that the report has yet to be submitted for NRC staff review and approval. As part of this response, address how the EPRI report may be applied if the NRC staff's RG 1.147 activities establish conditions on the use ASME Code Case N-889 (e.g., limiting the maximum fluence or displacements-per-atom (dpa) level for which these CGR methods may be used), or determine the Code Case to be unacceptable.

Response to RAI 04:

The EPRI report "Models of Irradiation-Assisted Stress Corrosion Cracking of Austenitic Stainless Steels in Light Water Reactor Environments," (TR Reference 46) was developed to provide crack growth rate models for high fluence austenitic stainless steels. The technical justification for the crack growth equations is contained in Volume 1 of the EPRI report (TR Reference 46) and included several Expert Panel reviews by EPRI. The Expert Panel reviewed and ranked IASCC data and provided guidance on the use of the data for modeling, developing disposition curves, reporting, to address IASCC issues. Experimental issues, as well as gaps in understanding them are discussed in the EPRI report (TR Reference 46) and were considered in the development of the crack growth rate guidance.

For fluence levels greater than $5x10^{20}$ n/cm^2 (E>1MeV) up to and including $3x10^{21}$ n/cm^2 (E>1MeV), the BWR HWC crack growth equation specified in Equation 2-11 of the EPRI report (TR Reference 46) is also specified in WCAP-17096, Rev. 3 [2]. This is the same as to the CGR equation specified in revision 2 of WCAP-17096-NP-A [3] for the same fluence range, therefore the use of this equation was previously justified and approved by the Staff.

For fluence ranges greater than $3x10^{21}$ n/cm^2 (E>1MeV), WCAP-17096, Revision 3 states:

[1] The list of items on TR page 2-6 is missing CE-ID Item 5.2; this item is specified as potentially subject to fluence greater than $3x10^{21}$ n/cm^2 (E>1MeV). The NRC staff is currently judging this as a document quality issue rather than a technical one. Please check all of your documents to make sure these kinds of errors are fixed.

- *"For weld locations subjected to fluence levels above 3x10^{21} n/cm^2 (E>1MeV), a CGR model appropriate for the fluence level and material must be used and a justification for its use provided. One appropriate model is available in EPRI Report Models of Irradiation-Assisted Stress Corrosion Cracking of Austenitic Stainless Steels in Light Water Reactor Environments, Volume 2: Disposition Curves Application [46], which provides the technical basis for ASME Code Case N-889 [52]."*

Section 2 of Volume 2 of [7] the EPRI report (TR Reference 46) provides additional information for the technical justification of the crack growth rate models calibrated to laboratory IASCC data, compares them with previously approved IASCC models, and discusses the uncertainties and variability of the calibration data relative to the models. The data used for model calibration were screened by an Expert Panel and had an average expert ranking <3 (the better data). Two models were calibrated, for BWR NWC and low-ECP environments, where low-ECP includes both BWR HWC and PWR primary water environments. Additional details on the models and how they were developed are discussed in Section 5 of Volume 1 of [7] the EPRI report (TR Reference 46). It is also noted that the PWR water chemistry CGR equation specified in [7] the EPRI report (TR Reference 46) is conservative compared to the CGR equation for fluence levels greater than 5x10^{20} n/cm^2 (E>1MeV), up to and including 3x10^{21} n/cm^2 (E>1MeV).

The ASME BPVC published Code Case N-889 [24] in July 12, 2018, which contains the latest reference stress corrosion cracking growth rate curves for Irradiated Austenitic Stainless Steel in Light-Water Reactor Environments (both PWR and BWR environments are considered). The technical bases of this Code Case are per ASME Code Committee Record 17-506 [15], along with Reference 46 of the TR. The NRC issued the conditional acceptance of this Code Case in Draft RG 1.147 Revision 20 (ADAMS Accession No. ML20120A631) [25], which was issued for public comment. The NRC specified several conditions for the use of Code Case N-889. The final NRC approval of this Code Case is scheduled for the end of 2021. Based on the conditional approval of N-889 and the NRC comments in Draft RG 1.147 Rev. 20, that the following revisions to WCAP-17096, Revision 3 will be made:

For fluence levels less than or equal to 5x10^{20} n/cm^2 (E>1MeV) (0.75 dpa):

- For weld locations subjected to fluence levels less than or equal to 5x10^{20} n/cm^2 (E>1MeV) (0.75 dpa), the maximum of the crack growth rate in 2017 ASME Section XI Appendix C-8520 **or** the crack growth equation in ASME Code Case N-889 (TR Reference 52) will be used. The conditions for the use of Code Case N-889 identified in Draft Regulatory Guide 1.147, Rev. 20 [25] will be met.

For fluence levels greater than 5x10^{20} n/cm^2 (E>1MeV) (0.75 dpa), up to and including 1.4x10^{22} n/cm^2 (E>1MeV) (20 dpa):

- For weld locations subjected to fluence levels greater than 5x10^{20} n/cm^2 (E>1MeV) (0.75 dpa) up to and including 1.4x10^{22} n/cm^2 (E>1MeV) (20 dpa), the crack growth equation in ASME Code Case N-889 (TR Reference 52) will be used. The conditions for the use of Code Case N-889 identified in Draft Regulatory Guide RG 1.147, Rev. 20 [25] will be met for the development of acceptance criteria.

For fluence ranges greater than 1.4x10^{22} n/cm^2 (E>1MeV) (20 dpa):

- For weld locations subjected to fluence levels greater than 1.4x10^{22} n/cm^2 (E>1MeV) (20 dpa), the CGR model in EPRI Topical Report, "*Models of Irradiation-Assisted Stress Corrosion Cracking of Austenitic Stainless Steels in Light Water Reactor Environments, Volume 2:*

Disposition Curves Application," (TR Reference 46), will be used. The acceptance criteria developed for fluence levels greater than 1.4×10^{22} n/cm^2 (E>1MeV) (20 dpa) must be approved by the NRC as discussed in Draft RG 1.147, Rev. 20 [25].

If ASME Code Case N-889 [24] is approved by the NRC with no Conditions, the previous Conditions do not need to be met.

The above changes are shown in the following pages of the TR:

Section 2.2	Page 2-6	See page 2 of Attachment 2
Appendix C	Page C-9	See page 14 of Attachment 2

These same changes will be made to the following in the TR:

Appendix A	Page A-58
Appendix A	Page A-27
Appendix C	Page C-15
Appendix C	Page C-21
Appendix C	Page C-27
Appendix C	Page C-49
Appendix C	Page C-56
Appendix C	Page C-67
Appendix C	Page C-73
Appendix C	Page C-79
Appendix C	Page C-100
Appendix E	Page E-21
Appendix E	Page E-27
Appendix E	Page E-41
Appendix E	Page E-49
Appendix E	Page E-56

NRC RAI 05

Applicable TR Sections, Appendices, Tables, Figures, or Pages:
TR Section 2.2 (TR Pages 2-12 and 2-13) and TR Appendix C, CE-ID Item 3.2

Background and Issue:
TR Section 2.2 includes the following statement on Page 2-12 for CE "Expansion" category components within the scope of Condition 3 for Group 4 Components:

> *"Note that the licensee shall submit the plant-specific fracture mechanics analysis justifying detected flaws within one year after any inspection that detects relevant conditions as defined in Table 5-2 of MRP-227."*

The TR did not apply the above statement to the evaluation of CE core shroud ribs and rings ("Expansion" category component in TR Appendix C, CE-ID Item 3.2) based on the following statements in TR Section 2.2, Pages 2-12, and 2-13:

> *"Note that the CE component, 'Core Support Barrel*[2] *Assembly (Welded) - Ribs and Rings,' has been identified as an inaccessible component for MRP-227, Rev. 1 as discussed in Part C of RAI 12 [41]. This change to MRP-227, Rev. 1 was accepted by the NRC in the draft SE for MRP-227, Rev. 1 [57]. Since a fracture mechanics analysis is no longer applicable to this component, the associated submittal requirement as contained in WCAP-17096-NP-A, Rev. 2 was not included in the methodology defined herein.*"

The staff does not find component "inaccessibility" to be a valid basis for excluding the core shroud ribs and rings from needing submittal of a plant-specific analysis. The component-specific assessment needs for the core shroud rib and rings in TR Appendix C, CE-ID Item 3.2 now call for performance of a functionality analysis instead of a fracture mechanics analysis. However, neither TR Section 2.2 nor TR Appendix C, CE-ID Item 3.2 call for the component-specific functionality analysis of the ribs and rings to be submitted to the staff. The Condition 3, Group 4 criterion for submittal of a plant-specific analysis does not become null/void based on the components being inaccessible or changing the analysis type from a fracture mechanics analysis to a functionality analysis. To properly address the intent of Condition 3 for the core shroud ribs and rings, the TR may need to specify that functionality analysis of these components be submitted to the staff within one year of the linked "Primary" inspection of the core shroud plates if it triggers "Expansion" to the core shroud ribs and rings.

NRC Request:
Justify why TR Section 2.2 and TR Appendix C, CE-ID Item 3.2 do not include any criteria specifying the need for submitting a functionality analysis of the shroud ribs and rings (or a justified component-specific replacement schedule) to the NRC within one year of the linked "Primary" component inspection if it triggers Expansion to these components per Item C3 of Table 5-2 in MRP-227, Rev. 1-A. Otherwise, revise these TR sections to specify submittal of the functionality analysis or alternative component-specific replacement schedule for these components.

[2] "Ribs and Rings" are listed as subcomponents of the CE Core Shroud Assembly (Welded), not the CE Core Support Barrel Assembly. The NRC staff is currently judging this as a document quality issue rather than a technical one. Please check the TR to make sure these apparent errors may be fixed.

Response to RAI 05:

Yes, the methodology for the CE Core Shroud Assembly (Welded) - Ribs and Rings (CE-ID: 3.2) was changed from a fracture mechanics analysis to a functionality analysis. The submittal requirement had originally appeared to be a consequence of the fracture mechanics analysis, and was therefore deleted. The TR will be revised to reflect that the submittal requirement remains applicable to this component.

The text in the submittal requirement in Section 2.2 of the TR will be revised to state that it applies to the functionality analysis of the Ribs and Rings component. These changes are shown on page 3 of Attachment 2. The submittal requirement was added to CE-ID: 3.2 as shown on page 16 of Attachment 2.

NRC RAI 06

Applicable TR Sections, Appendices, Tables, Figures, or Pages:
TR Section 2.3 (TR Pages 2-14 – 2-15), TR Appendix C, CE-ID Items 6.1 and 6.2; and
TR Appendix E, W-ID Items 4.2 and 4.3

Background and Issue:
EPRI Letter No. MRP 2019-023 (Sept. 2019, Ref. 8) incorporates EPRI Letter No. MRP 2019-009 (Ref. 9). MRP 2019-009 establishes "Needed" Interim Guidance for performing one-time VT-3 visual inspections of middle axial welds (MAWs) and lower axial welds (LAWs) in WEC core barrel assemblies or CE core support barrel (CSB) assemblies.

The staff noted that none of the above TR sections reference the existence of the EPRI Letters MRP 2019-009 and MRP 2019-023. Further, the assessments in TR Appendix C, CE-ID Items 6.1 and 6.2; and TR Appendix E, W-ID Items 4.2 and 4.3 do not provide any acceptance criteria or data analysis methods for the one-time VT-3 visual examination that must be performed for the MAWs and LAWs per the guidelines in EPRI Letter No. MRP 2019-009.

NRC Request:
(a) Justify why TR Section 2.3 does not incorporate the above 2019 Interim Guidance for WEC core barrel assembly and CE core support assembly MAWs and LAWs, as provided in EPRI Letter Nos. MRP 2019-023 and MRP 2019-009.
(b) Justify why the acceptance criteria and data analysis methods TR Appendix C, CE-ID Items 6.1 and 6.2 and *TR* Appendix E, W-ID Items 4.2 and 4.3 do not include acceptance criteria and data analysis methods for the one-time VT-3 inspection of the MAWs and LAWs for WEC core barrels and CE core support barrels.

Response to RAI 06:

The interim guidance in MRP 2019-009 [9] recommends the use of EVT-1, eddy current, or ultrasonic examinations of the middle axial weld (MAW) and lower axial weld (LAW) for CE and Westinghouse NSSS plants when performing the Primary MRP-227 EVT-1 inspections of the girth welds. Note that MRP 2019-009 does not specify VT-3 as discussed in the RAI due to the expected type of potential degradation.

Additionally, the guidance in MRP 2019-009 is classified as NEI 03-08 "Good Practice," which is at the discretion of the licensee to implement, rather than NEI 03-08 "Needed" guidance that must be implemented.

For reference, Section 5.8 of NEI 03-08 [17] defines three classifications of guidance:

"• *Mandatory – to be implemented at all plants where applicable*
• *Needed – to be implemented whenever possible but alternate approaches are acceptable*
• *Good Practice – implementation is expected to provide significant operational and reliability benefits, but the extent of use is at the discretion of the individual plant/utility."*

The purpose of MRP-227, Rev. 1-A [6] is to define the examination requirements (Section 4) and examination acceptance criteria (Section 5), whereas the purpose of this TR is to establish

methodologies and data requirements for use in the disposition of MRP-227 examination results that are outside of the examination acceptance criteria in Section 5 of MRP-227. MRP 2019-009 defines "Good Practice" examination requirements. As such, they are not equivalent to the "Needed" requirements of the tables in Section 4 and Section 5 of MRP-227, Rev. 1-A. Furthermore, since the MRP 2019-009 items are examination requirements, they are not directly relevant to the acceptance criteria methodologies and data requirements of the TR.

Therefore, it follows that it is not necessary to include MRP 2019-009 in the discussion of industry activity in Section 2.3 of the TR, as discussed in Part (a) of this RAI. Likewise, it was not necessary to cite MRP 2019-009 in the CE methodologies in Appendix C (CE-ID Items 6.1 and 6.2), nor in the Westinghouse methodologies in Appendix E (W-ID Items 4.2 and 4.3) as discussed in Part (b) of this RAI.

NRC RAI 07

Applicable TR Sections, Appendices, Tables, Figures, or Pages:

TR Section 3.3 (TR Page 3-2), TR Appendix C, CE-ID Items 1 and 1.2; TR Appendix E, W-ID Items 6 and 6.1.

Background:

TR Section 3.3 addresses the general basis for evaluation of bolting failures in "bingo chart" style bolted assemblies where acceptance may be determined based on patterns of failed and unfailed bolts. The second paragraph of TR Section 3.3 states the following:

> *"The use of acceptable bolting patterns as acceptance criteria that allow individual bolt failures is established in the industry. The PWROG (from prior Westinghouse Owners Group efforts) has developed acceptable bolting patterns for baffle-to-former bolting for Westinghouse reactor internals designs* ***[first emphasized statement]****. The plant-specific applicability of these generic acceptable patterns must be confirmed by the user* ***[second emphasized statement]****, since certain inputs may have changed over the two decades since these patterns were originally developed. The PWROG has supported development of similar strategies for core barrel bolt inspections in Babcock & Wilcox (B&W) plants. The methodology for performing an acceptable bolting pattern or similar strategy is beyond the scope of the current task."* (Emphasis Added)

Issue:

The first emphasized statement in the above paragraph is not consistent with component evaluation methods specified in TR Appendix C for CE bolted core shroud assemblies, core shroud bolts (CE-ID Item 1) and barrel-shroud bolts (CE-ID Item 1.2). These CE-ID items cite the original acceptable bolting pattern analysis (ABPA) methodology in WCAP-15029-P-A, Rev. 1 (1999). WCAP-15029-P-A, Rev. 1 is identified as an ABPA methodology for baffle-former bolting and barrel-former bolting in WEC-designed reactors. The WCAP does not provide methods for performing ABPAs for core shroud bolting and barrel-shroud bolting in CE-designed reactors.

NRC Request:

(a) Explain how CE-ID Items 1 and 1.2 are eligible to use the methodology in WCAP-15029-P-A, Rev. 1. If it can be adequately demonstrated that WCAP-15029-P-A, Rev. 1 is an acceptable ABPA method for these CE components, then please revise TR Section 3.3 to state this. Otherwise, please delete the citation of the WCAP-15029-P-A for these CE items, and replace it with a more generic method for bolted core support structures, such as ASME Section III, Subsection NG and applicable ASME Section III Appendices, Code Cases, etc.

Issue:

The NRC-approved APBA methodology in WCAP-15029-P-A, Rev. 1 (1999) has recently been updated in PWROG-18034-P, Revision 0, "Updates to the Methodology in WCAP-15029-P-A, Rev. 1, 'Westinghouse Methodology for Evaluating the Acceptability of Baffle-Former-Barrel Bolting Distributions Under Faulted Load Conditions'," October 2018 (Proprietary, Reference 11 for Non-Proprietary version). In its May 26, 2020 SE (Reference 12) for PWROG-18034-P, the NRC staff concluded that the methodology of PWROG-18034-P may be used to supplement the methodology of WCAP-15029-P-A Rev. 1 for developing ABPAs for WEC-designed reactors, subject to limitations in Section 5.0 of the SE.

NRC Request:

(b) Identify whether PWROG-18034-P is intended for use as a supplement to WCAP-15029-P-A, Rev. 1 for performing ABPAs to meet evaluation criteria in TR Appendix E, W-ID Items 6 and 6.1. If so, please supplement W-ID Items 6 and 6.1 and the references in TR Section 7 to include the PWROG-18034-P methodology.

Issue:

The second emphasized statement indicates that WEC plants that apply the generic acceptable bolting pattern analysis (APBA) methods to the analytical evaluation of degraded bolting patterns per their plant design bases "must" confirm the applicability of the generic acceptable bolting patterns. In contrast, Section 1.1 of the TR states that there "are no specific NEI 03-08 Mandatory, Needed, or Good Practice elements" in the TR. The PWROG's inclusion of the word "must" in the 3rd sentence of the 2nd paragraph in TR Section 3.3 indicates that this is either a "Mandatory" or "Needed" action for any WEC PWR that will be applying the generic APBA methodology to the assessment of failed, degraded, or non-degraded bolting in the plant design.

NRC Request:

(c) Clarify whether the action associated with the 3rd sentence in the 2nd paragraph in TR Section 3.3 is being established as an NEI 03-08 "Mandatory" or "Needed" activity for those licensees or LRA/SLRA applicants that will need to apply ABPA methods to the analytical evaluations of degraded RVI bolting patterns per their plant-specific design bases.

Response to RAI 07:

(a) The applicability of the baffle-former bolt evaluation methodologies in WCAP-15029-P-A [18] and WCAP-15030-NP-A [19] is for Westinghouse NSSS plants. Only two CE NSSS plant designs have the bolted baffle-former assembly design similar to the Westinghouse NSSS plant designs: Fort Calhoun and Palisades. Fort Calhoun shutdown in October 2016, and Palisades is scheduled to shutdown in the Spring of 2022. Therefore, the methodologies in CE-ID: 1 and 1.2 will not be utilized in the future, and CE-ID: 1 and CE-ID: 1.2 will be removed from the TR. In addition, CE-ID: 1.1 for the Core Support Column Bolts will be removed, although WCAP-15029-P-A was not included, since this methodology would only apply to NSSS plant designs with bolt baffle-former assemblies.

    These changes are shown by the removal of CE-ID: 1, CE-ID: 1.1, and CE-ID: 1.2 from the list of CE components on page 7 of Attachment 2, in the removal of CE-ID: 1 for Core Shroud Bolts on pages 8 and 9 of Attachment 2, in the removal of CE-ID: 1.1 for Core Support Column Bolts on pages 10 and 11 of Attachment 2, and also in the removal of CE-ID: 1.2 for Barrel-Shroud Bolts on pages 12 and 13 of Attachment 2. The flowcharts for CE-ID: 1, CE-ID: 1.1, and CE-ID: 1.2 were removed as shown on pages 18-20.

(b) PWROG-18034-P-A supplements WCAP-15029-P-A and PWROG-18034-NP-A supplements WCAP-15030-NP-A. However, WCAP-17096, Rev. 3 [2] was transmitted for NRC review in July 2019, which was one year before PWROG-18034-P-A [20] and PWROG-18034-NP-A [21] were issued in July 2020. Therefore, the NRC approved versions of PWROG-18034 could not be incorporated into W-ID: 6 and W-ID: 6.1.

A statement will be added to W-ID: 6 for Baffle-Former Bolts, that PWROG-18034-P-A [20] and PWROG-18034-NP-A [21] supplement the methodology (WCAP-15029-P-A [18] and WCAP-15030-NP-A [19]) as shown on page 22 of Attachment 2.

This same change will also be made for W-ID: 6.1 for Barrel-Former Bolts, as shown on pages 23 and 24 of Attachment 2. PWROG-18034-P-A and PWROG-18034-NP-A will also be added as references as shown on page 6 of Attachment 2.

(c) Westinghouse performed generic ABPAs for Westinghouse 2-loop, 3-loop, and 4-loop NSSS plants over twenty years ago. The generic ABPAs were based on bounding loading and operating parameters for each category of plant. Certain parameters used in these evaluations have changed over that timeframe. Therefore, it is necessary for each licensee to confirm that the ABPAs remain applicable if they intend to apply the generic results to their plant.

The 3rd sentence in the 2nd paragraph in Section 3.3 of the TR was included to identify that the applicability of the generic ABPA evaluations must be confirmed, if those results are to be applied to on a plant specific basis. As discussed in Section 1.1 of this TR, "There are no specific NEI 03-08 Mandatory, Needed, or Good Practice elements in this document." Therefore, this sentence is not intended to invoke an NEI 03-08 "Needed" requirement.

NRC RAI 08

Applicable TR Sections, Appendices, Tables, Figures, or Pages:
TR Section 3.6 (TR Page 3-3) and TR Appendix C, CE-ID Items 7, 8 and 9

Background and Issue:
TR Section 3.6 states that the three CE "Primary" components identified below may be evaluated through the use of plant-specific fatigue analysis to address potential for fatigue crack formation. The components are:

- CSB flexure weld located in the CSB assembly (TR CE-ID Item 7)
- Core support plate in the lower support structure (TR CE-ID Item 8)
- Fuel alignment plate in the upper internals assembly (TR CE-ID Item 9)

MRP-227, Rev. 1-A, Table 4-2, Line Item Nos. "C7," "C9[3]," and "C10[3]" (corresponding to TR CE-ID Items 7, 8, and 9, respectively) specifies EVT-1 inspections of these components if a plant-specific evaluation cannot demonstrate that the components screen out for fatigue.

In addition, for CE-ID Item 7, MRP-227, Rev. 1-A states that, "*If the CSB Flexure Weld screens out for fatigue, SCC must still be considered. This can be accomplished by performing an evaluation using plant-specific or bounding information, or by performing the inspection as prescribed above*."

TR Section 3.6 is silent on the need for a plant-specific or bounding SCC evaluation of the CSB flexure weld, in addition to a plant-specific fatigue evaluation, as an alternative to performing the EVT-1 exam of this item. (Note that this is specified in CE-ID Item 7.)

Further, the fatigue screening criteria for all three of these components assumes that the current licensing basis (CLB) for the plant will include the needed cyclical loading analyses for fatigue screening in order to demonstrate that no fatigue crack will be formed. It is not clear how a fatigue screening is to be performed if the CLB for the specified component does not include the cyclical loading analysis needed to achieve the fatigue screening objective.

NRC Request:
(a) Provide a justification for not including in TR Section 3.6, the need for a plant-specific or bounding SCC evaluation of the CSB flexure weld, in addition to a plant-specific fatigue evaluation, as an alternative to performing the EVT-1 exam of this item.

(b) Provide a justification for not including under CE-ID Item 7 to include a plant-specific or bounding SCC evaluation (if this weld screens out for fatigue) as an alternative to performing the EVT-1 exam.

(c) Address how component-specific fatigue screening will be achieved by analysis for all three components specified above if the CLB for the component does not include a cyclical loading analysis to support fatigue screening. Otherwise, modify the recommended evaluation criteria in the above TR sections.

[3] The NRC staff is currently judging this numbering issue as a document quality issue rather than a technical one. Please check the TR to make sure these apparent errors are fixed.

Response to RAI 08:

Background:

It is acknowledged that the numbering in MRP-227, Rev. 1-A [6] for CE components 9, 10, 11, 11.1 and 12 is not consistent with the TR. This was done because the CE components were re-numbered in the TR due to relocating the Lower Support Structure – Core Support Column component to Expansion; however, this same re-numbering was not applied in MRP-227, Rev. 1-A. The TR will be revised to use the component numbering in MRP-227, Rev. 1-A as follows:

- CE-ID: 9 Lower Support Structure – Core Support Plate
- CE-ID: 10 Upper Internals Assembly – Fuel Alignment Plate
- CE-ID: 11 Control Element Assembly – Instrument Guide Tubes
- CE-ID: 11.1 Control Element Assembly – Remaining Instrument Guide Tubes
- CE-ID: 12 Lower Support Structure – Deep Beams

These changes are shown on page 7 of Attachment 2 for the list of CE component methodologies in Appendix C, and the same changes as shown above will be made to the component numbering throughout the TR.

(a) Additional clarification will be added to Section 3.6 of the TR regarding the SCC evaluation of the CSB Flexure Weld. The Examination requirements in Table 4-2 of MRP-227, Rev. 1-A [6] states that an SCC evaluation is necessary, as well as the fatigue screening in order to conclude that an EVT-1 inspection is not necessary:

> *"If screening for fatigue cannot be satisfied by plant-specific evaluation, perform enhanced visual (EVT-1) examination, no later than 2 refueling outages from the beginning of the license renewal period. Subsequent examination on a ten-year interval.*
>
> *If the CSB Flexure Weld screens out for fatigue, SCC must still be considered. This can be accomplished by performing an evaluation using plant-specific or bounding information, or by performing the inspection as prescribed above."*

A statement will be added to Section 3.6 to provide this clarification as shown on page 4 of Attachment 2.

(b) The requested clarification is contained in the Inspection Method section of CE-ID: 7 of the TR:

> *"If screening for fatigue cannot be satisfied by plant-specific evaluation, visual examination is required. SCC must be considered either by performing a plant-specific (bounding) evaluation or otherwise by performing the visual examination. See MRP-227 Table 4-2."*

The intent of this requirement is that both a fatigue screening and SCC evaluation must be performed; otherwise, an EVT-1 inspection must be performed. A statement, "Regardless of the ability to screen out for fatigue" will be added before the 2nd sentence above, to further clarify this intent. This additional statement that will be included in CE-ID: 7 is shown on page 17 of Attachment 2.

(c) The design basis loading including the cyclic loading and number of cycles are defined for reactor internals components; therefore, the needed inputs are available to perform a fatigue screening. The results of the fatigue screening evaluation will be provided to the licensees, to determine the impact

on the plant licensing basis. The potential to update the plant licensing basis with the fatigue screening results, versus the outage impact of performing the EVT-1 inspections as part of the aging management plan for these components will be considered by the utility.

NRC RAI 10

Applicable TR Sections, Appendices, Tables, Figures, or Pages:
TR Appendix C, CE-ID Items 2, 2.1, 3, 3.1, 5.2, 5.3, 6, 6.1, 6.2, and 11;
TR Appendix E, W-ID Items 3.1, 3.2, 4, 4.2, and 4.3

Background and Issue:
The TR items listed above include tables providing recommended pre-inspection and flaw analysis methods for these "Primary" or "Expansion" category components. The TR recommends a lower bound fracture toughness value of 34.6 ksi-√inch for LEFM analysis of SS base metal or weld components with neutron fluence exposures that are projected to exceed 1 x 10^{22} n/cm^2 (E > 1.0 MeV; > 15 dpa) at the end of the service period of interest. The TR does not provide a basis for citation of the 34.6 ksi-√inch lower bound LEFM fracture toughness value or cite a reference for this value.

NRC Request:
Provide a justification for the use of the lower bound LEFM fracture toughness value of 34.6 ksi-√inch for SS base metal or weld components with projected neutron fluence exposures in excess of 1 x 10^{22} n/cm^2 (E > 1.0 MeV; > 15 dpa) at the end of the service period of interest.

Response to RAI 10:

MRP-211, Revision 1 [22] (Materials Reliability Program: PWR Internals Age-Related Material Properties, Degradation Mechanisms, Models, and Basis Data – State of Knowledge) was issued in 2017. MRP-211 discusses the current state of knowledge, available relevant data, and technical bases for trend model formulations of key aging mechanisms for long term functionality evaluations, including fracture toughness. MRP-211 concluded that all fracture toughness data are bounded by a saturated value for K_{JC} of 34.6 ksi-√in. MRP-211 further states in the last paragraph of subsection 2.2.4: "this value provides an appropriate lower bound of all fracture toughness data and is recommended by the expert panel for use in engineering evaluations and assessments." It is also noted that the use of the fracture toughness of 34.6 ksi-√inch for fluence values exceeding 1 x 10^{22} n/cm^2 (E > 1.0 MeV) is consistent with WCAP-17096-NP-A, Revision 2 which was approved by the NRC.

The TR will be revised to include MRP-211 as a reference as shown on page 6 of Attachment 2, and note 2 will be added to the tables that include the 34.6 ksi-√inch fracture toughness values, as shown on page 15 of Attachment 2 for CE-ID: 2.

This same change will be made to the following in the TR:

Appendix C	Page C-17
Appendix C	Page C-23
Appendix C	Page C-29
Appendix C	Page C-52
Appendix C	Page C-58
Appendix C	Page C-69
Appendix C	Page C-75
Appendix C	Page C-81
Appendix C	Page C-101
Appendix E	Page E-23

Appendix E Page E-30
Appendix E Page E-43
Appendix E Page E-52
Appendix E Page E-59

NRC RAI 11

Applicable TR Sections, Appendices, Tables, Figures, or Pages:
TR Section 3.5, "Visual Other"

Background and Issue:
TR Section 3.5 discusses use of visual inspection techniques for the detection of distortion due to void swelling in PWR RVI components and methods for evaluating such aging effects. The discussion emphasizes that swelling-related distortion may occur in the gap areas of CE core shrouds that are designed in two vertical sections. The TR states that the evaluation of swelling in the gap areas of these types of core shrouds "*will require additional sensitivity studies to relate the swelling level to the predicted distortion and gap opening in the structure*." The TR also states that the sensitivity studies may be necessary "*to evaluate the current condition, predict future effects, and determine aging management strategies, which are beyond the scope of*" the TR.

To date, there has been little industry-reported operating experience with indications of changes in dimension, swelling, or distortion in WEC, CE, or B&W RVI components. Further, the specific acceptance criteria and data analysis methods for evaluation of these detected aging effects is apparently outside the scope of the TR.

NRC Request:
Clarify whether the intent of TR Section 3.5 is for swelling-related distortion assessments to be the responsibility of licensees through implementation of component-specific analysis and acceptance criteria for these detected aging effects.

Response to RAI 11:

As discussed in the RAI, Section 3.5 of the TR states that evaluation of this swelling-related data will require additional sensitivity studies to relate the swelling level to the predicted distortion and gap openings in the structure. The sensitivities studies were performed for some plants and are documented in MRP-230 [23] (TR reference 6). A licensee utilizing these sensitivity studies must confirm that they are applicable to their plant.

CE-ID: 4a discusses the methodology for the evaluation of welded core shroud assemblies with two vertical sections. The methodology states: "a plant-specific plan should be developed for evaluating and mitigating the potential relevant conditions." These plant-specific analyses will be performed by the licensee.

Text will be added to Section 3.5 of the TR that swelling-related distortion evaluations will be performed by the licensee. The text is shown on page 4 of Attachment 2.

NRC RAI 14

Applicable TR Sections, Appendices, Tables, Figures, or Pages:
TR Appendix E, W-ID Item 1 (Page E-3).

Background and Issue:
The TR includes the following "Data Requirements" for WEC control rod guide tube (CRGT) assembly guide cards:

- Guide card innermost hole ligament wear depth (or remaining ligament thickness) or slot opening width if ligament thickness is greater than 100 percent worn away.
- Continuous guidance member wear if ligament at the first guide card above the continuous is projected to wear-through before the time of the next inspection.
- *Guide card innermost hole ligament wear depth (or remaining ligament thickness) or slot opening width if ligament thickness is greater than 100 percent worn away.*

The NRC staff noted that the first bullet is repeated under "Data Requirements" on Page E-3 of the TR.

In WCAP-17096-NP-A, Rev. 2, the CRGT assembly guide card methods include a data requirement for guide tube operational effective full power years (EFPYs) at the time of the inspection or measurement. This data requirement is not included in the TR[4].

NRC Request:
Provide an explanation for the noted discrepancies between WCAP-17096-NP-A, Rev. 2 and Rev. 3 for the "Data requirements" associated with W-ID Item 1.

Response to RAI 14:

Yes, there is a typographical error in the data requirements section of W-ID: 1 for the Guide Plates (Cards). The revised bulleted items for the "Data Requirements" section is shown on page 21 of Attachment 2.

[4] The NRC staff is currently judging this issue as a document quality issue rather than a technical one. Please check the TR to make sure these apparent errors are fixed.

References

1. NRC NUREG-2191, Volume 2, "Generic Aging Lessons Learned for Subsequent License Renewal Applications," Chapter XI.M16A, *PWR Vessel Internals*, 2017 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML16274A399).

2. Westinghouse Report, WCAP-17096-NP, Revision 3, "Reactor Internals Acceptance Criteria Methodology and Data Requirements," July 2019 (ADAMS Accession No. ML19218A179).

3. Westinghouse Report, WCAP-17096-NP-A, Revision 2 "Reactor Internals Acceptance Criteria Methodology and Data Requirements," August 2016 (ADAMS Accession No. ML16279A320).

4. NRC Correspondence Letter and Safety Evaluation from Mr. Kevin Hseuh (USNRC) to Ms. Anne Demma (EPRI), "Final Safety Evaluation of WCAP-17096-NP, Revision 2, 'Reactor Internals Acceptance Criteria Methodology and Data Requirements,'" May 3, 2016 (ADAMS Accession No. ML16061A243).

5. *Materials Reliability Program: Pressurized Water Reactor Internals Inspection and Evaluation Guidelines (MRP-227-A)*. EPRI, Palo Alto, CA: 2011. 1022863. (ADAMS Accession Nos. ML12017A194, ML12017A196, ML12017A197, ML12017A191, ML12017A192, ML12017A195 and ML12017A199).

6. *Materials Reliability Program: Pressurized Water Reactor Internals Inspection and Evaluation Guidelines (MRP-227, Revision 1-A).* EPRI, Palo Alto, CA: 2019. 3002017168. (ADAMS Accession No. ML19339G350).

7. *Models of Irradiation-Assisted Stress Corrosion Cracking of Austenitic Stainless Steels in Light Water Reactor Environments, Volume 2: Disposition Curves Application*. EPRI, Palo Alto, CA, 2014. 3002003103.

8. EPRI Letter No. MRP 2019-023, "Transmittal of MRP-227-A-Related Interim Guidance Regarding PWR Core Barrel," September 3, 2019 (ADAMS Accession ML19249B102; includes enclosure of MRP 2019-009 [RAI Set Ref. 9 below]).

9. EPRI Letter No. MRP-2019-009, "Transmittal of NEI 03-08 "Good Practice" Interim Guidance Regarding MRP-227-A and MRP-227, Revision 1 Core Barrel and Core Support Barrel Inspection Requirements," July 17, 2019.

10. NRC Correspondence Letter and Safety Evaluation from Mr. Dennis Morey (USNRC) to Mr. Brian Burgos (EPRI), "Final Safety Evaluation for Electric Power Research Institute Topical Report MRP-227, Revision 1, 'Materials Reliability Program: Pressurized Water Reactor Internals Inspection and Evaluation Guideline,'"April 25, 2019 (ADAMS Accession No. ML19081A001).

11. Westinghouse Report, PWROG-18034-NP, Revision 0, "Updates to the Methodology in WCAP-15030-NP-A, Rev. 0, 'Westinghouse Methodology for Evaluating the Acceptability of Baffle-Former-Barrel Bolting Distributions Under Faulted Load Conditions,'" October 2018 (ADAMS Accession No. ML18306A491).

12. Final Safety Evaluation by the Office of Nuclear Reactor Regulation, Topical Report PWROG-18034-P, Revision 0, "Updates to the Methodology in WCAP-15030-NP-A, Rev. 0, 'Westinghouse Methodology for Evaluating the Acceptability of Baffle-Former-Barrel Bolting Distributions Under Faulted Load Conditions,'" May 26, 2020 (ADAMS Accession No. ML20134M356).

13. Westinghouse Technical Bulletin TB-19-5, "Westinghouse NSSS PWR Thermal Shield Degradation," Westinghouse Non-Proprietary Class 3, October 9, 2019 (ADAMS Accession No. ML19302F228).
14. Westinghouse Document, Presentation to the NRC Staff (Non-Proprietary Version), "Westinghouse Baffle-Former Bolt Predictive Methodology," Josh McKinley and Matt Palamara, September 18, 2019 (ADAMS Accession No. ML19274F472).
15. ASME Code Committee Record #17-506 - Irradiation Assisted Stress Corrosion Crack Growth in Irradiated Materials.
16. U.S. Nuclear Regulatory Commission Letter, "U.S. NRC Request for Additional Information for Pressurized Water Reactor Owner's Group Topical Report, WCAP-17096-NP, Revision 3, 'Reactor Internals Acceptance Criteria Methodology and Data Requirements'," September 21, 2020.
17. Nuclear Energy Institute Document, NEI 03-08, Rev. 4, "Guideline for the Management of Materials Issues," October 2020.
18. Westinghouse Report, WCAP-15029-P-A, Rev. 1, "Westinghouse Methodology for Evaluating the Acceptability of Baffle-Former-Barrel Bolting Distributions Under Faulted Load Conditions," January 1999.
19. Westinghouse Report, WCAP-15030-NP-A, Rev. 0, "Westinghouse Methodology for Evaluating the Acceptability of Baffle-Former-Barrel Bolting Distributions Under Faulted Load Conditions," January 1999.
20. PWROG Owner's Group Report, PWROG-18034-P-A, Rev. 0, "Updates to the Methodology in WCAP-15029-P-A, Rev. 1, 'Westinghouse Methodology for Evaluating the Acceptability of Baffle-Former-Barrel Bolting Distributions Under Faulted Load Conditions'," July 2020.
21. PWROG Owner's Group Report, PWROG-18034-NP-A, Rev. 0, "Updates to the Methodology in WCAP-15030-NP-A, Rev. 0, 'Westinghouse Methodology for Evaluating the Acceptability of Baffle-Former-Barrel bolting Distributions Under Faulted Load Conditions'," July 2020."
22. *Materials Reliability Program: PWR Internals Age-Related Material Properties, Degradation Mechanisms, Models, and Basis Data – State of Knowledge (MRP-211, Revision 1).* EPRI Palo Alto, CA: 2017. 3002010270.
23. *Materials Reliability Program: Functionality Analysis for Westinghouse and Combustion Engineering Representative PWR Internals (MRP-230 Rev. 2).* EPRI, Palo Alto, CA: 2012. 1021026.
24. ASME Code Case N-889, "Reference Stress Corrosion Crack Growth Rate Curves for Irradiated Austenitic Stainless Steel in Light-Water Reactor Environments," July 12, 2018.
25. Proposed Revision 20 to Regulatory Guide RG 1.147, "Inservice Inspection Code Case Acceptability, ASME Section XI, Division 1," January, 2021. *(Attached in PRIME)*

ATTACHMENT 2

Markups to WCAP-17096-NP, Rev. 3 that Reflect the Changes Discussed in the RAI Reponses

Purpose

This attachment contains the redline markups of the changes to WCAP-17096-NP, Rev. 3 [2] that are discussed in the responses to the U.S. Nuclear Regulatory Commission (NRC) Requests for Additional Information (RAIs) [1]. These changes to the TR will be incorporated into the NRC-approved version that will be issued after the Final Safety Evaluation is issued.

References

1. U.S. Nuclear Regulatory Commission Letter, "U.S. NRC Request for Additional Information for Pressurized Water Reactor Owners Group Topical Report, WCAP-17096-NP, Revision 3, 'Reactor Internals Acceptance Criteria Methodology and Data Requirements'," September 21, 2020.
2. Westinghouse Report, WCAP-17096-NP, Rev. 3, "Reactor Internals Acceptance Criteria Methodology and Data Requirements," July 2019 (ADAMS Accession No. ML19218A179).

Condition 2 (B&W, CE, and Westinghouse RVI Components)

> "*For welds or components with neutron fluences greater than* $3x10^{21}$ *n/cm*^{2}*, the CGR models used in plant-specific analyses must be either previously approved by the NRC-staff, or the basis for the model must be submitted along with the analysis.*"

Condition 2 Resolution

An appropriate CGR model for high fluence welds has been developed by EPRI [46]. This model is included in ASME Code Case N-889 [52]. The CGR models at the other fluence values are also consistent with ASME Code Case N-889 [52] identified below. Therefore, the following discussion has been included for the applicable flaw tolerance evaluations in WCAP-17096, Rev. 3.

" • For weld locations subjected to fluence levels less than or equal to $5x10^{20}$ n/cm^2 (E>1MeV) (0.75 dpa), the maximum of the crack growth rate in 2017 ASME Section XI Appendix C-8520 or the PWR crack growth equation per ASME Code Case N-889 [52] will be used. The conditions for the use of Code Case N-889 identified in Draft Regulatory Guide RG 1.147, Rev. 20 [67] will be met.

- For weld locations subjected to fluence levels greater than $5x10^{20}$ n/cm^2 (E>1MeV) (0.75 dpa) up to and including $1.4x10^{22}$ n/cm^2 (E>1MeV) (20 dpa), the crack growth equation in ASME Code Case N-889 [52] will be used. The conditions for the use of Code Case N-889 identified in Draft Regulatory Guide RG 1.147, Rev. 20 [67] will also be met for the development of acceptance criteria.

- For weld locations subjected to fluence levels greater than $1.4x10^{22}$ n/cm^2 (E>1MeV) (20 dpa), the CGR model in EPRI Topical Report, "*Models of Irradiation-Assisted Stress Corrosion Cracking of Austenitic Stainless Steels in Light Water Reactor Environments, Volume 2: Disposition Curves Application,*" [46], will be used. The acceptance criteria developed for fluence levels greater than $1.4x10^{22}$ n/cm^2 (E>1MeV) (20 dpa) must be approved by the NRC as discussed in Draft RG 1.147, Rev. 20 [67].

  If ASME Code Case N-889 [52] is approved by the NRC with no Conditions, the previous Conditions do not need to be met."

This statement has been added to the following components:

- B&W-ID: A.1.9, Baffle plates
- B&W-ID: A.2.11, Lower Grid Rib Section
- CE-ID: 2, Core Shroud Assembly (Welded) – Core Shroud Plate-Former Plate Weld
- CE-ID: 2.1, Core Shroud Assembly (Welded) – Remaining Axial Welds
- CE-ID: 3, Core Shroud Assembly (Welded) – Shroud Plates
- CE-ID: 3.1, Core Shroud Assembly (Welded) – Remaining Axial Welds
- CE-ID: 5.3, Core Support Barrel Assembly – Upper Axial Weld
- CE-ID: 6, Core Support Barrel Assembly – Middle Girth Weld
- CE-ID: 6.1, Core Support Barrel Assembly – Middle Axial Weld
- CE-ID: 6.2, Core Support Barrel Assembly – Lower Axial Weld
- W-ID: 3.1, Core Barrel Assembly – Upper Girth Weld
- W-ID: 3.2, Core Barrel Assembly – Upper Axial Weld
- W-ID: 4, Core Barrel Assembly – Lower Girth Weld

The following statement has been added to Group 4 RVI components without expansion criteria:

> *Note that the licensee shall submit the plant-specific fracture mechanics analysis justifying the detected flaws within one year after any inspection that detects relevant conditions as defined in Table 5-2 of MRP-227.*

This statement has been added to the following components:

- CE-ID: 2.1, Core Shroud Assembly (Welded) – Remaining Axial Welds
- CE-ID: 3.1, Core Shroud Assembly (Welded) – Remaining Axial Welds

The CE component, "Core Shroud Assembly (Welded) – Ribs and Rings," has been identified as an inaccessible component for MRP-227, Rev. 1 as discussed in Part C of RAI 12 [41]. This change to MRP-227, Rev. 1 was accepted by the NRC in the draft SE for MRP-227, Rev. 1 [57]. Note that the submittal requirement as contained in WCAP-17096-NP-A, Rev. 2 remains applicable, and is included in the methodology as clarified below:

> *Note that the licensee shall submit the plant-specific functionality analysis within one year after any inspection that detects relevant conditions as defined in Table 5-2 of MRP-227.*

This statement has been added to the following component:

- CE-ID: 3.2, Core Shroud Assembly (Welded) – Ribs and Rings

Group 5 RVI Components

The following statement has been added to W-ID: 2, "Control Rod Guide Tube Assembly – Lower Flange Weld," Group 5 RVI component:

> *Note that plant-specific analysis of acceptable intact weld patterns for this component must be submitted to the NRC to determine if review and approval is needed, within one year after any inspection that triggers the expansion criteria in Table 5-3 of MRP-227.*

Note that this condition would also be expected to apply to W-ID: 2.1, "Control Rod Guide Tube Assembly – Remaining Lower Flange Welds," and was therefore also added to this component.

The following statement has been added to W-ID: 9, "Thermal Shield Assembly – Thermal Shield Flexures," Group 5 RVI component:

> *Note that plant-specific analysis of acceptable dynamic response of the remaining flexures for this component must be submitted to the NRC to determine if review and approval is needed, within one year after any inspection that detects relevant conditions as defined in the "Examination Acceptance Criteria" column in Table 5-3 of MRP-227.*

General fracture mechanics procedures for calculating critical flaw sizes and growth rates are described within the I&E Guidelines. In order to apply these procedures, the appropriate irradiation history, loading conditions and stress intensity solutions must be identified. These factors are all dependent on both the flaw location and the plant design.

This task outlines specific fracture mechanics analysis requirements for each of the visual EVT-1 examinations included in the Primary and Expansion tables of the I&E Guidelines.

3.5 VISUAL OTHER

A VT-1 level examination was identified to examine potential swelling-related distortion in some welded core shroud structures originally designed by CE. The intention of the examination is to provide semi-quantitative data that can be used to evaluate the overall level of swelling in the structure. The original engineering evaluation and assessments (i.e., functionality analysis), which is known to be conservative, predicted large gap openings at specific locations in the shroud structure.

These examinations are meant to provide an early warning of swelling in the structure. A functionality evaluation was performed in MRP-230 [6], which indicates that gap openings may indicate void swelling. Evaluation of this swelling-related data will require additional sensitivity studies to relate the swelling level to the predicted distortion and gap opening in the structure. Sensitivity studies may be necessary to evaluate the current condition, predict future effects, and determine aging management strategies, which are beyond the scope of this document. Additionally, a plant-specific plan should be developed for evaluating and mitigating potential void swelling.

3.6 FATIGUE (QUALIFY BY EVALUATION)

Three component items in the CE design are included in the list of Primary components due to concerns about fatigue and permit the use of a "fatigue analysis" to address potential fatigue degradation[1]. Due to the plant-specific nature of the evaluation required for license extension programs, fatigue analysis was not included in the MRP engineering evaluation and assessments (i.e., functionality analysis). It is considered a high probability that an evaluation will demonstrate a negligible probability of fatigue crack initiation in these components. Note that an additional requirement exists for the CSB Flexure Weld, as discussed in the Examination Method/Frequency of Table 4-2 of MRP-227, Rev. 1-A [36], to evaluate SCC, if this component screens out for fatigue. However, pending resolution by evaluation[2], these component items are included in the Primary Component list.

[1] The referenced three component items are: 1) the core support plate, 2) the fuel alignment plate, and 3) the core support barrel flexure weld.

[2] Note that a fatigue analysis performed as a part of a time-limiting aging analysis (TLAA) fulfills this requirement.

25. EPRI Letter, "Proposed Edits to WCAP-17096-A Draft," August 5, 2013. (ADAMS Accession No. ML13219A183)

26. EPRI Letter, "Proposed Edits to WCAP-17096-A-Draft," January 16, 2014. (ADAMS Accession No. ML14041A042)

27. EPRI Letter, "Proposed Edits to WCAP-17096-NP, Revision 2," April 10, 2014. (ADAMS Accession No. ML14104B579)

28. EPRI Letter, "Transmittal of Revised Text for Draft WCAP-17096-NP Revision 2 for NRC Review, LTR-RIAM-12-159, Revision 2," July 8, 2014. (ADAMS Accession No. ML14191A014)

29. NRC Letter, "Request for Additional Information Related to WCAP-17096-NP, Revision 2, 'Reactor Internals Acceptance Criteria Methodology and Data Requirements'," August 18, 2014. ADAMS Accession No. ML14177A071)

30. EPRI Letter, "Transmittal of the Additional RAI Responses to the NRC relating to WCAP-17096-NP, Revision 2, 'Reactor Internals Acceptance Criteria Methodology and Data Requirements'," October 31, 2014. (ADAMS Accession No. ML14308A076)

31. NRC Letter, Request for Additional Information Related to WCAP-17096-NP, Revision 2, "Reactor Internals Acceptance Criteria Methodology and Data Requirements," February 11, 2015. ADAMS Accession No. ML15005A052)

32. EPRI Letter, "Transmittal of Responses (PWROG-15035-NP, Revision 0) to the NRC Request for Additional Information (RAI) on PWR Owners Group (PWROG) WCAP-17096-NP, Revision 2, 'Reactor Internals Acceptance Criteria Methodology and Data Requirements'," May 5, 2015. (ADAMS Accession No. ML15128A198)

33. PWROG Letter OG-09-290, Rev. 0, "Transmittal of Draft Report WCAP-17096, Rev 0 'Reactor Internals Acceptance Criteria Methodology and Data Requirements,' PA-MSC-0473," July 27, 2009.

34. NRC Letter, "Final Safety Evaluation of WCAP-17096-NP, Revision 2, 'Reactor Internals Acceptance Criteria Methodology and Data Requirements (TAC No. ME4200)'," May 3, 2016. (ADAMS Accession No. ML16061A243)

35. PWROG Letter, OG-16-299, "Transmittal of NRC Approved Topical Report WCAP-17096-NP-A, Rev. 2 'Reactor Internals Acceptance Criteria Methodology and Data Requirements,' (TAC Number ME4200), PA-MSC-0473," September 2, 2016. (ADAMS Accession Nos. ML16279A319 and ML16279A320)

36. *Materials Reliability Program: Pressurized Water Reactor Internals Inspection and Evaluation Guidelines (MRP-227, Revision 1-A)*. EPRI, Palo Alto, CA: 2019. 3002017168.

60. PWROG Letter, OG-18-46, Transmittal of Approved "Needed" Interim Guidance for Addressing Accelerated Guide Card Wear Issue Described in NSAL-17-1, (LTR-RIDA-17-270, Revision 0), PA-MSC-1471," February 20, 2018.

61. Westinghouse Report, WCAP-17451-P, Rev. 1, "Reactor Internals Guide Tube Wear – Westinghouse Domestic Fleet Operational Projections," October 2013.

62. *BWRVIP-158: BWR Vessel and Internals Project, Flaw Proximity Rules for Assessment of BWR Internals.* EPRI, Palo Alto, CA: 2006. 1014387.

63. EPRI Letter, BWRVIP 2019-016, "White Paper on Suggested Content for PFM Submittals to the NRC," February 27, 2019.

64. PWR Owner's Group Report, PWROG-18034-P-A, Rev. 0, "Updates to the Methodology in WCAP-15029-P-A, Rev. 1, 'Westinghouse Methodology for Evaluating the Acceptability of Baffle-Former-Barrel Bolting Distributions Under Faulted Load Conditions'," July 2020.

65. PWR Owner's Group Report, PWROG-18034-NP-A, Rev. 0, "Updates to the Methodology in WCAP-15030-NP-A, Rev. 0, 'Westinghouse Methodology for Evaluating the Acceptability of Baffle-Former Barrel Bolting Distributions Under Faulted Load Conditions'," July 2020.

66. *Materials Reliability Program: PWR Internals Age-Related Material Properties, Degradation Mechanisms, Models, and Basis Data – State of Knowledge (MRP-211, Revision 1).* EPRI Palo Alto, CA: 2017. 3002010270.

67. Proposed Revision 20 to Regulatory Guide RG 1.147, "Inservice Inspection Code Case Acceptability, ASME Section XI, Division 1," January, 2021.

APPENDIX C
ACCEPTANCE CRITERIA METHODOLOGY AND DATA REQUIREMENTS FOR COMBUSTION ENGINEERING PRIMARY AND EXPANSION COMPONENT ITEMS

CE Primary and Expansion Components

~~CE-ID: 1 Core Shroud Assembly (Bolted) – Core Shroud Bolts~~

- ~~CE-ID: 1.1 Core Shroud Assembly (Bolted) – Core Support Column Bolts~~
- ~~CE-ID: 1.2 Core Shroud Assembly (Bolted) – Barrel-Shroud Bolts~~

CE-ID: 2 Core Shroud Assembly (Welded) - Core Shroud Plate-Former Plate Weld

- CE-ID: 2.1 Core Shroud Assembly (Welded) – Remaining Axial Welds

CE-ID: 3 Core Shroud Assembly (Welded) – Shroud Plates

- CE-ID: 3.1 Core Shroud Assembly (Welded) – Remaining Axial Welds
- CE-ID: 3.2 Core Shroud Assembly (Welded) – Ribs and Rings

CE-ID: 4 Core Shroud Assembly (Bolted) – Assembly
CE-ID: 4a Core Shroud Assembly (Welded) – Assembly
CE-ID: 5 Core Support Barrel Assembly – Upper Flange Weld

- CE-ID: 5.1 Core Support Barrel Assembly – Lower Girth Weld
- CE-ID: 5.2 Core Support Barrel Assembly – Upper Girth Weld
- CE-ID: 5.3 Core Support Barrel Assembly – Upper Axial Weld
- CE-ID: 5.4 Lower Support Structure – Lower Core Support Beams

CE-ID: 6 Core Support Barrel Assembly – Middle Girth Weld

- CE-ID: 6.1 Core Support Barrel Assembly – Middle Axial Weld
- CE-ID: 6.2 Core Support Barrel Assembly – Lower Axial Weld
- CE-ID: 6.3 Lower Support Structure – Core Support Columns

CE-ID: 7 Core Support Barrel Assembly – CSB Flexure Weld
CE-ID: 9 Lower Support Structure – Core Support Plate
CE-ID: 10 Upper Internals Assembly – Fuel Alignment Plate
CE-ID: 11 Control Element Assembly –Instrument Guide Tubes

- CE-ID: 11.1 Control Element Assembly – Remaining Instrument Guide Tubes

CE-ID: 12 Lower Support Structure – Deep Beams

~~CE-ID: 1~~	**~~Core Shroud Assembly (Bolted)~~**
	~~Core Shroud Bolts~~
~~Category:~~	~~Primary~~ ~~Applicability:~~ ~~Bolted plant designs~~
~~Degradation Effect:~~	~~See MRP-227 Table 4-2~~
~~Expansion Link:~~	~~See MRP-227 Table 4-2~~
~~Function:~~	~~The shroud-former bolts fasten the shroud plates to the barrel-former structure.~~
~~Inspection~~	
~~Method:~~	~~Volumetric examination. See MRP-227 Table 4-2.~~
~~Coverage:~~	~~See MRP-227 Table 4-2~~
~~Observable Effect:~~	~~Volumetric examination should reliably detect flaws greater than 30% through-shaft cracking.~~
~~Failure~~	
~~Failure Mechanism:~~	~~Known IASCC cracking of similar highly irradiated bolts has been reported.~~
~~Failure Effect:~~	~~Loss of structural stability~~
~~Failure Criteria:~~	~~Require an acceptable bolting pattern~~
~~Methodology~~	
~~Goal:~~	~~Must demonstrate that projected number of additional bolt failures will not threaten acceptable pattern prior to next scheduled inspection.~~
~~Data Requirements:~~	~~Loads~~ ~~Bolting patterns~~ ~~Shroud design~~ ~~Fast neutron (dpa) distribution in core shroud~~ ~~Projected bolt failure rate~~
~~Analysis:~~	~~The observed pattern of failed bolts must be consistent with the pre-defined acceptable bolt pattern or be analyzed in real time, and have a reasonable margin to protect against additional failures during the inspection interval.~~

~~**CE-ID: 1**~~ ~~**Core Shroud Assembly (Bolted)**~~

~~**Core Shroud Bolts**~~

~~Acceptance Criteria:~~

~~Procedures for establishing acceptable bolting patterns for the baffle-to-former bolts in Westinghouse-designed plants have been established in [13] and [14]. This methodology has been reviewed and accepted by the NRC is a Safety Evaluation in 1998 (TAC No. MA1152). The same methodology should be applied to the two operating CE plants with bolted core shrouds.~~

1. ~~Observed pattern of unfailed bolts meets pre-defined acceptance criteria or is analyzed in real time for acceptability.~~
2. ~~Less than 50% of initial margin consumed.~~

~~$N_f < (N - N_{req})/2$ (See below)~~

~~The margin is defined in terms of the number of intact bolts beyond the number required for the acceptable bolting pattern. The margin, M, at any time is simply:~~

~~$M = N - N_{req} - N_f$~~

~~Where~~

~~N = total number of shroud-former bolts~~

~~N_{req} = number of shroud-former bolts in acceptable pattern~~

~~N_f = number of failed bolts.~~

~~Assuming that there are no failed bolts at the beginning of life, the initial margin is simply: $(N - N_{req})$. N_{req} is determined via an Acceptable Bolting Analysis. For operation through the next 10-year interval, require that no more than 50% of initial margin be consumed at the time of the inspection. Require that no more than 50% of initial margin be consumed for any subsequent inspection as well.~~

~~Approach:~~ ~~No generic effort required. Only one plant is affected.~~

~~Additional Action(s):~~ ~~None~~

~~CE-ID: 1.1~~	**~~Core Shroud Assembly (Bolted)~~**		
	~~Core Support Column Bolts~~		
~~Category:~~	~~Expansion~~	~~Applicability:~~	~~Bolted plant designs~~
~~Degradation Effect:~~	~~See MRP-227 Table 4-5~~		
~~Primary Link:~~	~~See MRP-227 Table 4-5~~		
~~Function:~~	~~Attach core support columns to core support plate.~~		
~~Inspection~~			
~~Method:~~	~~Volumetric examination. See MRP-227 Table 4-5.~~		
~~Coverage:~~	~~See MRP-227 Table 4-5~~		
~~Observable Effect:~~	~~Volumetric examination should reliably detect flaws greater than 30% through-shaft cracking.~~		
~~Failure~~			
~~Failure Mechanism:~~	~~IASCC, fatigue, and IE~~		
~~Failure Effect:~~	~~Loss of structural stability~~		
~~Failure Criteria:~~	~~Determine acceptable number of support columns required to maintain structural integrity.~~		
~~Methodology~~			
~~Goal:~~	~~Conservatively assume that failure of a bolt results in loss of attachment between the support column and the core support plate. Establish functional requirements for core support columns.~~ ~~a. During normal operation system of support columns should resist core plate deformation due to mechanical or thermal loading. Core plate requirements for "flatness" and fuel assembly alignment.~~ ~~b. During limiting accident transient system must maintain structural integrity.~~		
~~Data Requirements:~~	~~Loads on core support plate.~~ ~~Displacement tolerances on lower core plate.~~ ~~Fluence accumulated by the (core support column bolts / lower support column bolts).~~ ~~Constitutive model for stainless steel properties as a function of irradiation.~~		
~~Analysis:~~	~~See associated figures in MRP-227.~~		

~~**CE-ID: 1.1**~~	~~**Core Shroud Assembly (Bolted)**~~
	~~**Core Support Column Bolts**~~ ~~1. Structural model of relevant portions of lower support structure.~~ ~~2. Build FEA model of lower support structure that includes support columns and core support plate. Model should be capable of removing individual column or breaking attachment to lower core support plate. Would require multiple iterations to establish "acceptable patterns" of core support columns and support column bolts.~~ ~~3. Structural model must be run for functional requirements A and B.~~ ~~4. Determine margin for additional failures.~~ ~~Assume number of failures in next 10 years is equal to number observed to date.~~ ~~N = # of support columns~~ ~~N_f = # of observed flawed columns~~ ~~N_{req} = # of columns in relevant acceptable pattern~~ ~~Margin = $N-N_{req}$~~
~~Acceptance Criteria:~~	~~Require that no more of 1/2 of columns in margin are failed:~~ ~~Nf < (N-Nreq)/2~~
~~Approach:~~	~~Generic program to share first-of-a-kind effort.~~ ~~Pilot analysis of lower support structure to identify critical issues.~~ ~~Expect final acceptance based on plant-specific analysis.~~
~~Additional Action(s):~~	~~Note that the licensee shall submit the plant-specific analysis for the acceptable distribution of intact components within one year after any inspection that triggers the expansion criteria as defined in Table 5-2 of MRP-227.~~

~~CE-ID: 1.2~~	**~~Core Shroud Assembly (Bolted)~~** **~~Barrel-Shroud Bolts~~**		
~~Category:~~	~~Expansion~~	~~Applicability:~~	~~Bolted plant designs~~
~~Degradation Effect:~~	~~See MRP-227 Table 4-5~~		
~~Primary Link:~~	~~See MRP-227 Table 4-5~~		
~~Function:~~	~~Maintain structural integrity of barrel-shroud structure.~~		
~~Inspection~~			
~~Method:~~	~~Volumetric examination. See MRP-227 Table 4-5.~~		
~~Coverage:~~	~~See MRP-227 Table 4-5~~		
~~Observable Effect~~	~~Volumetric examination should reliably detect flaws greater than 30% through-shaft cracking.~~		
~~Failure~~			
~~Failure Mechanism:~~	~~Cracking by combined effects of IASCC and fatigue.~~		
~~Failure Effect:~~	~~Inability to maintain structural stability~~		
~~Failure Criteria:~~	~~Require an acceptable bolting pattern.~~		
~~Methodology~~			
~~Goal:~~	~~Must demonstrate an acceptable bolting pattern.~~		
~~Data Requirements:~~	~~Loads~~ ~~Bolting patterns~~ ~~Shroud design~~ ~~Fast neutron (dpa) distribution in core shroud~~ ~~Projected bolt failure rate~~ ~~Acceptable bolting pattern analysis~~		

~~**CE-ID: 1.2**~~ ~~**Core Shroud Assembly (Bolted)**~~

~~**Barrel-Shroud Bolts**~~

~~Analysis:~~ ~~The observed pattern of failed bolts must meet the pre-defined acceptable bolt pattern and have a reasonable margin to protect against additional failures during the inspection interval. The margin is defined in terms of the number of intact bolts beyond the number required for the acceptable bolting pattern. The margin, M, at any time is simply:~~

~~M = N - Nreq - Nf~~

~~Where~~

~~N = total number of barrel-former bolts~~

~~N_{req} = number of barrel-former bolts in acceptable pattern~~

~~N_f = number of failed bolts.~~

~~Assuming that there are no failed bolts at the beginning of life, the initial margin is simply: (N-N_{req}). N_{req} is determined via an Acceptable Bolting Analysis. For operation through the next 10-year interval, require that no more than 50% of initial margin be consumed at the time of the inspection. Require that no more than 50% of initial margin be consumed for any subsequent inspection as well.~~

~~Acceptance Criteria:~~ ~~Procedures for establishing acceptable bolting patterns or the barrel-to-former bolts in Westinghouse designed plants have been established in [14]. This methodology has been reviewed an accepted by the NRC in a Safety Evaluation in 1998 (TAC No. MA1152). The same methodology should be applied to the two operating CE plant with bolted core shrouds.~~

1. ~~Observed pattern of unfailed bolts meets pre-defined acceptance criteria.~~

~~Approach:~~ ~~No generic effort required. Only one plant is affected.~~

~~Additional Action(s):~~ ~~None~~

CE-ID: 2 **Core Shroud Assembly (Welded)**

Core Shroud Plate-Former Plate Weld

- For weld locations subjected to fluence levels less than or equal to $5x10^{20}$ n/cm^2 (E>1MeV) (0.75 dpa), the maximum of the crack growth rate in 2017 ASME Section XI Appendix C-8520 **or** the crack growth equation in ASME Code Case N-889 [52] will be used. The conditions for the use of Code Case N-889 identified in Draft Regulatory Guide RG 1.147, Rev. 20 [67] will be met.
- For weld locations subjected to fluence levels greater than $5x10^{20}$ n/cm^2 (E>1MeV) (0.75 dpa) up to and including $1.4x10^{22}$ n/cm^2 (E>1MeV) (20 dpa), the crack growth equation in ASME Code Case N-889 [52] will be used. The conditions for the use of Code Case N-889 identified in Draft Regulatory Guide RG 1.147, Rev. 20 [67] will be met for the development of acceptance criteria.
- For weld locations subjected to fluence levels greater than $1.4x10^{22}$ n/cm^2 (E>1MeV) (20 dpa), the CGR model in EPRI Topical Report, "*Models of Irradiation-Assisted Stress Corrosion Cracking of Austenitic Stainless Steels in Light Water Reactor Environments, Volume 2: Disposition Curves Application* [46], will be used. The acceptance criteria developed for fluence levels greater than $1.4x10^{22}$ n/cm^2 (E>1MeV) (20 dpa) must be approved by the NRC as discussed in Draft RG 1.147, Rev. 20 [67].

  If ASME Code Case N-889 [52] is approved by the NRC with no Conditions, the previous Conditions do not need to be met.

Acceptance criteria can be developed for the entire operating license of a given plant by using predicted end-of-license fluence values. However, if there are changes to these fluence projections, such as in the event of a power uprate or change in core loading pattern, it would be necessary to confirm that the inputs selected based on fluence, such as IASCC growth rate and fracture toughness, remain applicable until the end of the operating license.

Failure

Failure Mechanism:	Cracking (IASCC)
Failure Effect:	1. Damage to peripheral fuel assemblies. 2. Through-wall crack provides leak path through shroud.
Failure Criteria:	An existing flaw is unacceptable if the flaw length projected at the next inspection cycle is beyond the allowable crack length. . No observable damage in corresponding sections of peripheral fuel assemblies.

Methodology

Goal:	Demonstrate that the cracking mechanism will not result in growth beyond the allowable crack length over the planned inspection interval.

CE-ID: 2 **Core Shroud Assembly (Welded)**

Core Shroud Plate-Former Plate Weld

applied, which may be either K dependent or K independent. In order to apply the acceptance criteria to a period beyond one refueling cycle (not to exceed a 10-year re-inspection interval), follow-up action is required to verify the assumptions used in the predicted crack-growth rate. A re-inspection of the indication at a future specified outage, for example, would provide data that could be used to satisfy this verification requirement. Evaluate impact of fatigue crack growth on observed flaw. Failure of the welds is assumed to occur when unstable crack growth is initiated from the analyzed flaw. Three options are outlined for determining the limiting allowable flaw length, based on neutron dose. Analysis methods are suggested for both pre-inspection and generic analysis (Suggested Pre-Inspection Analysis) and for flaws observed in-service (Suggested Flaw Specific Analysis), where more detailed characteristics of the flaw and its location are known. In all cases, a more detailed evaluation may be completed using a semi-elliptic surface flaw, but such an evaluation would require more detailed inspection by UT.

Fluence Range (n/cm^2 E>1MeV)	Dose (dpa)	Suggested Pre-Inspection Analysis	Suggested Flaw Specific Analysis
$\leq 3 \times 10^{20}$	≤ 0.5	LEFM using 150 ksi√in for fracture toughness(1) or Limit Load	Limit Load
$> 3\times10^{20}$ and $\leq 3\times10^{21}$	> 0.5 and ≤ 5	LEFM using 112 ksi√in for fracture toughness(1) or EPFM	EPFM
$> 3\times10^{21}$ and $\leq 1\times10^{22}$	> 5 and ≤ 15	LEFM using 50 ksi√in for fracture toughness(1)	LEFM using 50 ksi√in for fracture toughness(1)
$>1\times10^{22}$	> 15	LEFM using 34.6 ksi√in for fracture toughness(1,2)	LEFM using 34.6 ksi√in for fracture toughness(1,2)

Note:

1. Alternate fracture toughness values may be applied and justification for the alternate basis shall be submitted to the NRC for information within one year.
2. MRP-221 [66] provides the basis for this fracture toughness value.

Different evaluation options may be used depending upon the plant-specific fluence levels at the location of the weld being evaluated. Option 1, though conservative, can be used for all fluence levels.

CE-ID: 3.2	**Core Shroud Assembly (Welded)**
	Ribs and Rings
Approach:	Functionality analysis
Additional Action(s):	Note that the licensee shall submit the plant-specific functionality analysis within one year after any inspection that detects relevant conditions as defined in Table 5-2 of MRP-227.

CE-ID: 7	**Core Support Barrel Assembly**		
	CSB Flexure Weld		
Category:	Primary	Applicability:	All plants with welded core shrouds
Degradation Effect:	See MRP-227 Table 4-2		
Expansion Link:	See MRP-227 Table 4-2		
Function:	Primary core support structure		

Inspection

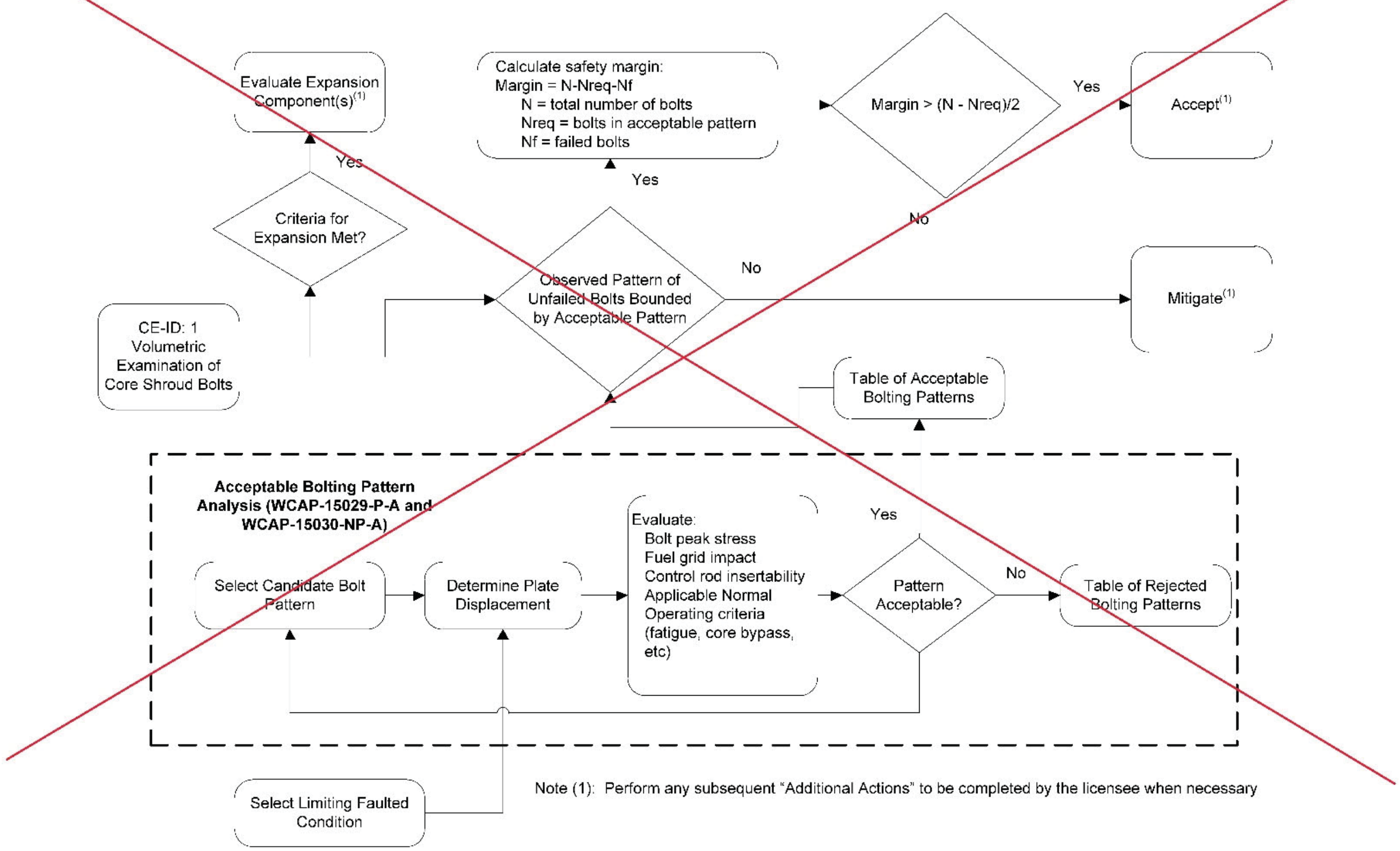
Method:	If screening for fatigue cannot be satisfied by plant-specific evaluation, visual examination is required. Regardless of the ability to screen out for fatigue, SCC must be considered either by performing a plant-specific (or bounding) evaluation, or otherwise by performing the visual examination. See MRP-227 Table 4-2.
Coverage:	See MRP-227 Table 4-2
Observable Effect:	The specific relevant condition is a detectable crack-like indication.

Inputs and Assumptions

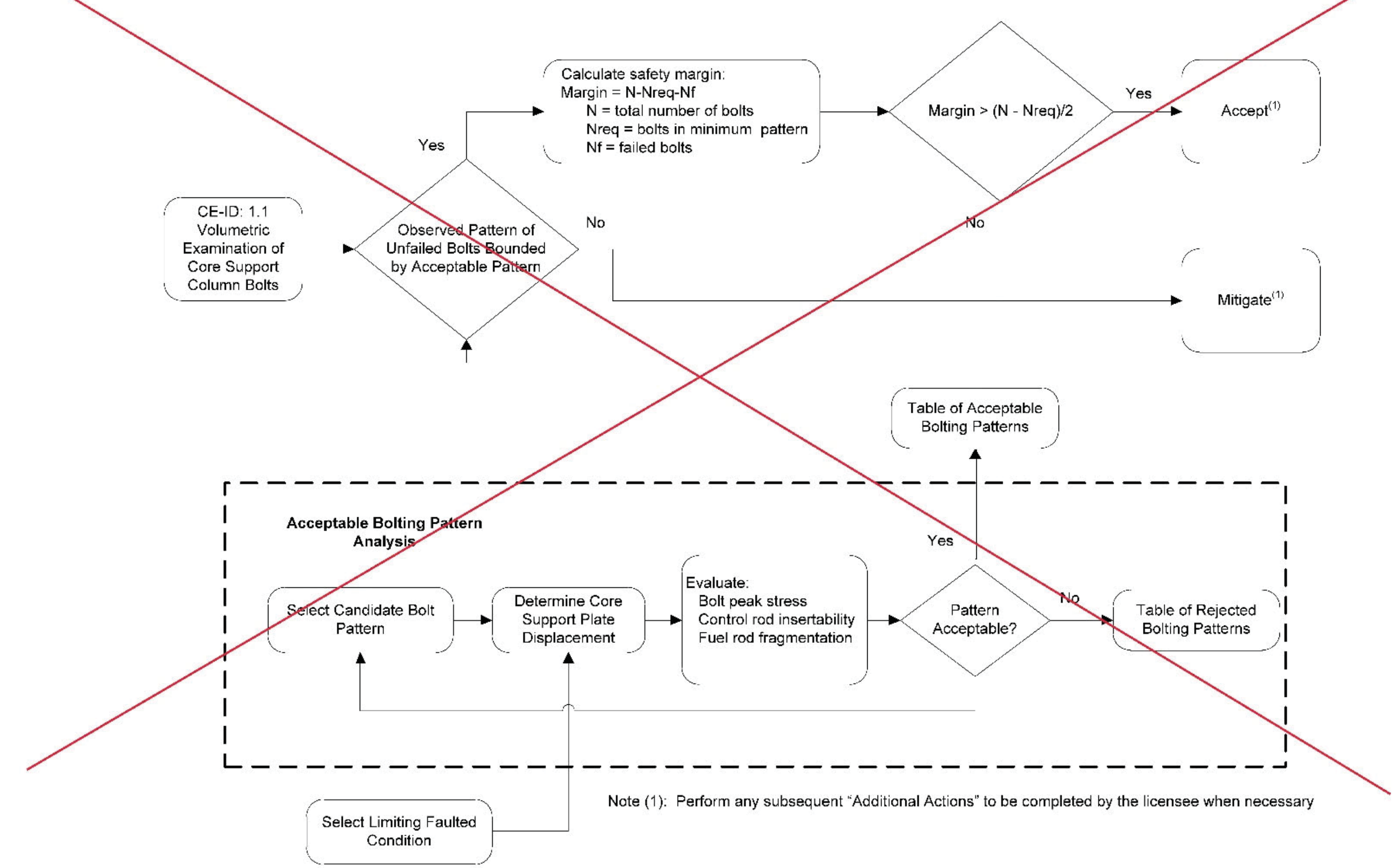
There are several inputs and assumptions that are critical to the development of acceptance criteria for the RV Internals CSB flexure weld. These items are stated below:

- The inspections identified in MRP-227 are intended to provide a sampling of potential locations of degradation. Under this approach, inspection of one side (surface) of the weld is intended to provide an adequate sampling for monitoring of fatigue.
- The change in resistance to fracture of the RV core barrel welds can be correlated to the accumulated fluence at each weld location. Welds that are subject to low fluence are considered to have a high degree of resistance to fracture. Correspondingly, those welds subject to high fluence have lower resistance to fracture.
- The prediction of crack growth is based on the stress intensity factor, K, calculated using linear elastic fracture mechanics. The rate of crack growth is dependent on the amount of neutron fluence that the weld is expected to accumulate over the licensed operating lifetime. Since there has been very limited experience of SCC initiated cracks in operating PWRs to date, growth rates developed for the prediction of SCC in BWRs are assumed to be appropriate for prediction of crack growth due to SCC in PWR reactor internals.
    - For weld locations subjected to fluence less than or equal to $5x10^{20}$ n/cm^2 (E>1MeV), the boiling water reactor (BWR) hydrogen water chemistry (HWC) crack growth equation specified in paragraph C-8520 of Appendix C of Section XI of the 2010 edition of the ASME Boiler and Pressure Vessel Code is appropriate. This crack growth rate model is consistent with the

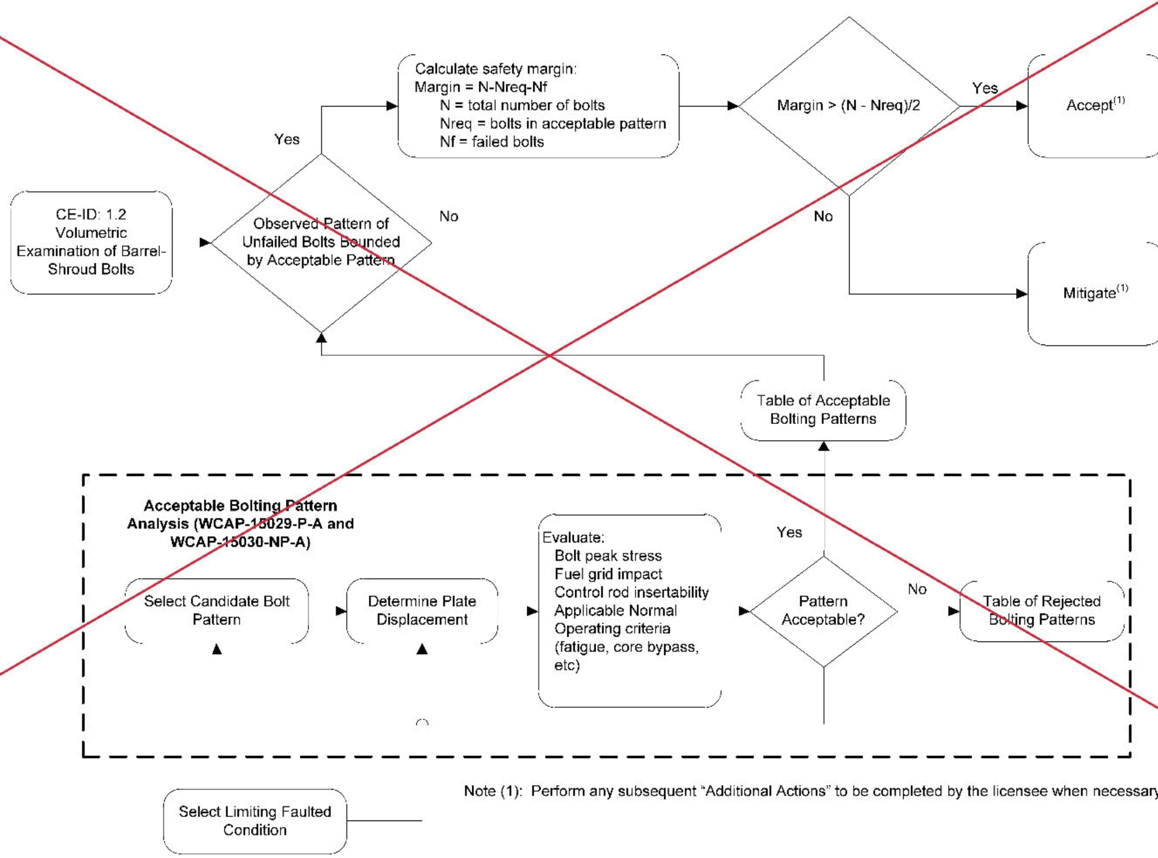
CE-ID: 1
Core Shroud Assembly (Bolted) - Core Shroud Bolts



CE-ID: 1.1
Core Shroud Assembly (Bolted) - Core Support Column Bolts



CE-ID: 1.2
Core Shroud Assembly (Bolted)- Barrel-Shroud Bolts



Note (1): Perform any subsequent "Additional Actions" to be completed by the licensee when necessary

W-ID: 1 **Control Rod Guide Tube Assembly**

Guide Plates (Cards)

Failure Effect: Wear of ligaments and opening of slots can cause loss of guidance and escape of the rodlet into the open center area of the guide tube. If enough consecutive guide cards lose guidance, the rodlet could plastically deform during accident conditions and prevent the rod cluster control assembly (RCCA) from inserting during a rod drop or during downward stepping. Wear-through along part of the length of a continuous guidance member could cause gouging and jamming of a rodlet and lead to prevention of insertion.

Failure Criteria: If more than six (6) consecutive guide cards have 100 percent rodlet diameter slot opening (loss of rodlet guidance), the guide tube position should not be relied on to perform a rod insertion function and that subsequent evaluation or corrective actions for that condition are beyond the scope of this criteria:

RCCA	Lower Guide Tube Length (in.)	Number of Open Cards
17x17	96	6
17x17	125	6
17x17	150	6
15x15	125	6
15x15	150	6
14x14	109, 119	6

The acceptance criteria for the guide card are provided in Table 1 and Table 2.

A guide tube must be removed from a rodded location before wear in the continuous guidance member section could impede rod insertion. This is ensured by removing a guide tube from service at least two (2) EFPY prior to reaching the limiting wear condition. The acceptance criteria for the continuous guidance member are defined in Table 4.

Methodology

Goal: Prevent operation of guide tubes with greater than the number of consecutive worn-through guide cards listed in the Failure Criteria section and with continuous guidance member wear as described in the Failure Criteria section.

Data Requirements:

- Guide card innermost hole ligament wear depth (or remaining ligament thickness) or slot opening width if ligament thickness is greater than 100 percent worn away
- Continuous guidance member wear if ligament at the first guide card above the continuous is projected to wear-through before the time of the next inspection
- Guide tube operational effective full power years (EFPYs) at the time of the inspection or measurement

W-ID: 6	**Baffle-former Assembly**	
	Baffle-former Bolts	
Category:	Primary Applicability: All plants	
Degradation Effect:	See MRP-227 Table 4-3	
Expansion Link:	See MRP-227 Table 4-3 and Interim Guidance MRP 2018-002 [48].	
Function:	The baffle-former bolts attach the baffle plates to the formers.	
Inspection		
Method:	Volumetric examination timing is dependent on plant configuration. See MRP 2016-021 [42] and MRP 2017-009 [43].	
Coverage:	See MRP-227 Table 4-3	
Observable Effect:	Indication of flaw from UT examination	
Failure		
Failure Mechanism:	Known IASCC cracking of similar highly irradiated bolts has been reported.	
Failure Effect:	Loss of structural stability	
Failure Criteria:	Require an acceptable bolting pattern	
Methodology		
Goal:	Must demonstrate that bolting patterns provide sufficient margin to acceptance criteria for continued operation through the re-inspection interval.	
Data Requirements:	Loads UT inspection results Test data for empirical model (as necessary) Baffle-former assembly design information including any applicable modifications Fast neutron (dpa) distribution in the baffle-barrel region (if needed)	
Analysis:	1. Structural integrity of the baffle-former assembly must be demonstrated by an evaluation of the bolting pattern. One approach to accomplish this is to perform an acceptable bolting pattern analysis (ABPA) per the methodology described in WCAP-15029-P-A [13] / WCAP-15030-NP-A [14] as supplemented by PWROG-18034-P-A [64] / PWROG-18034-NP-A [65], which have been reviewed and accepted by the NRC via Safety Evaluation. The bolting pattern considers functional bolts to include those with satisfactory UT inspection results or replacement bolts, as well as any portion of untestable bolts determined to be acceptable per Step 2. Otherwise, the un-inspectable bolts and those that show visual or UT indication of failure are treated as non-functional. This evaluation must consider any plant-specific variations of operating conditions from analysis assumptions, and may credit plant modifications such as upflow conversion to reduce bolt load.	

W-ID: 6.1	**Core Barrel Assembly** **Barrel-former Bolts**	
Category:	Expansion	Applicability: All plants
Degradation Effect:	See MRP-227 Table 4-6	
Primary Link:	See MRP-227 Table 4-6 and Interim Guidance MRP 2018-002 [48]	
Function:	Maintain structural integrity of baffle-former-barrel structure.	

Inspection

Method:	Volumetric examination. See MRP-227 Table 4-6
Coverage:	See MRP-227 Table 4-6
Observable Effect:	Indication of flaw from UT examination

Failure

Failure Mechanism:	Cracking Loss of bolt pre-load due to irradiation induced stress relaxation may exacerbate fatigue issue in aging plants
Failure Effect:	Potential for flow induced vibration due to loss of bolting constraint. Loss of structural stability
Failure Criteria:	Require an acceptable bolting pattern.

Methodology

Goal:	Must demonstrate an acceptable bolting pattern.
Data Requirements:	Loads/displacements Bolting patterns Baffle-former-barrel design Fast neutron (dpa) distribution in baffle-barrel region (if needed) Projected bolt failure rate
Analysis:	Procedures for establishing acceptable bolting patterns for the barrel-to-former bolts have been established in WCAP-15029-P-A [13] / WCAP-15030-NP-A [14] as supplemented by PWROG-18034-P-A [64] and PWROG-18034-NP-A [65]. This methodology has been reviewed and accepted by the NRC in a Safety Evaluation issued in 1998 (TAC No. MA1152).
Acceptance Criteria:	1. Observed pattern of unfailed bolts meets pre-defined acceptance criteria or is analyzed in real time. 2. Less than 50% of initial margin consumed. $N_f < (N - N_{req})/2$ (See below) The observed pattern of failed bolts must be shown to meet the conditions for being an acceptable bolt pattern per [13] / [14] as supplemented by [64] / [65] and have a reasonable margin to protect against additional failures during the inspection interval. The margin is defined in terms of the number of intact bolts

W-ID: 6.1 **Core Barrel Assembly**

Barrel-former Bolts

beyond the number required for the acceptable bolting pattern. The margin (M) at any time is simply:

$$M = N - N_{req} - N_f$$

Where
N = total number of barrel-former bolts
N_{req} = number of barrel-former bolts in acceptable pattern
N_f = number of failed bolts

Assuming that there are no failed bolts at the beginning of life, the initial margin is simply: ($N - N_{req}$). N_{req} is determined via an Acceptable Bolting Analysis, using the NRC-approved methodology outlined in [13] / [14] as supplemented by [64] / [65]. For operation through the next 10-year interval, require that no more than 50% of initial margin be consumed at the time of the inspection. Require that no more than 50% of initial margin be consumed for any subsequent inspection as well.

Approach: Generic work completed in previous PWROG program.

Plant-specific analysis is most often necessary to develop acceptable bolting patterns.

Additional Action(s): None