Materials Reliability Program: Pressurized Water Reactor Internals Inspection and Evaluation Guidelines (MRP-227, Revision 1-A)

2020 TECHNICAL REPORT

Materials Reliability Program: Pressurized Water Reactor Internals Inspection and Evaluation Guidelines (MRP-227, Revision 1-A)

All or a portion of the requirements of the EPRI Nuclear Quality Assurance Program apply to this product.

YES NO

EPRI Project Manager
K. Amberge

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3002017168
Final Report, June 2020

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THE FOLLOWING ORGANIZATIONS, UNDER CONTRACT TO EPRI, PREPARED THIS REPORT:

MRP and PWROG Joint Reactor Internals Planning Team

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NRC Safety Evaluation

In accordance with an NRC request, the NRC Safety Evaluation immediately follows this page. Other NRC and EPRI Material Reliability Program correspondence on this subject are included in the appendices.

December 2019 Note: The changes proposed by the NRC in the Safety Evaluation as well those proposed by the EPRI Materials Reliability Program in response to NRC Requests for Additional Information (RAIs) have been incorporated into the current version of the report (MRP-227 Revision 1-A).

June 2020 Note: During NRC acceptance reviews in early 2020, additional clarification questions and administrative issues related to the MRP-227 Revision 1-A topic report were identified by NRC staff reviewers (Ref. ML20006D152). These issues were addressed by EPRI and the industry team as detailed in the following pages. (Ref. ML20141L315).

Office of Nuclear Reactor Regulation
Topical Report Safety Evaluation

Topical Report Information	Review Information
Report Number: EPRI Letter MRP 2020-12 Title: None EPID: L-2020-TOP-0024 Docket No.: 99902016	Division/ Branch: DNLR/NVIB Project Manager: J. Holonich Reviewers: J. Medoff

Determination of Minimal Revisions

Is this the review of very limited scope?	Yes	X	No	
Does the TR change maintain the original SE conclusions?	Yes	X	No	
Do the staff methods for establishing the original conclusions remain unaffected?	Yes	X	No	

If any of the above questions are answered no, a simplified safety evaluation cannot be used.

Applicable Review Guidance Used

Electric Power Research Institute (EPRI) Letter No. MRP 2020-012 provides additional clarifications relative to the contents that are included in EPRI Report No. 3002170168 (also known as Technical Report MRP-227, Rev. 1-A, Materials Reliability Program: Pressurized Water Reactor Internals Inspection and Evaluations Guideline (Agencywide Documents Access and Management System (ADAMS) Accession No. ML19339G350).

Description of Topical Report Content (1000 Word Maximum)

MRP-227, Rev. 1-A provides the EPRI updated inspection and evaluation guidelines for pressurized water reactor (PWR) reactor vessel Internal components. The guidelines were approved in a Nuclear Regulatory Commission (NRC) staff-issued safety evaluation dated April 25, 2019 (ADAMS Accession No. ML19081A001) and endorsed by the staff in its correspondence letter to the EPRI MRP dated February 20, 2020 (ADAMS Accession No. ML20006D152).

The contents of the EPRI Letter No. MRP 2020-012 included some additional clarifications relative to the comments the NRC staff had made on the contents of TR MRP-227, Rev. 1-A, as issued in the February 20, 2020, letter. The comments were made relative to a few perceived component nomenclature disparities observed by the NRC staff in the MRP-227, Rev. 1-A report and some noted changes in tabular footnote designations between those made in the MRP-227, Rev. 1 version of the report and the corresponding footnote designations were included in the MRP-227, Rev. 1-A version of the report.

ML20141L315

Office of Nuclear Reactor Regulation
Topical Report Safety Evaluation

Topical Report Information	Review Information
Report Number: EPRI Letter MRP 2020-12 Title: None EPID: L-2020-TOP-0024 Docket No.: 99902016	Division/ Branch: DNLR/NVIB Project Manager: J. Holonich Reviewer: J. Medoff

Technical Evaluation

The staff has reviewed the contents of the EPRI Letter No. MRP 2020-012 submittal and has determined that it provides additional useful clarifications relative to the current contents of TR MRP-227, Rev.1-A. More specifically, the NRC staff has determined that EPRI Letter No. MRP 2020-012 achieves the following objectives relative to the contents of the MRP-227, Rev. 1-A report:

(1) Clarified the EPRI MRP's reasons for a few component nomenclature differences for some component descriptions that were included in the various tables of the MRP-227, Rev. 1-A report, in which the staff sought additional clarifications from the EPRI MRP. The staff found the EPRI MRP's explanations provided an acceptable basis for the differences in the component nomenclatures between tables in the report.

(2) Clarified that changes in specific footnote designations for specific TR table footnotes specified in the staff's February 20, 2020, were administrative in nature.

(3) Clarified that no further edits of the MRP-227, Rev. 1-A are necessary.

Office of Nuclear Reactor Regulation Topical Report Safety Evaluation	
Topical Report Information	**Review Information**
Report Number: EPRI Letter MRP 2020-12 Title: None EPID: L-2020-TOP-0024 Docket No.: 99902016	Division/ Branch: DNLR/NVIB Project Manager: J. Holonich Reviewers: J. Medoff

Conclusions

The staff considers the information and clarifications in EPRI's MRP 2020-012 letter to be an addendum or errata of the MRP-227, Rev. 1-A report. The staff finds the clarifications in the MRP 2020-012 letter to be reasonable and acceptable. Based on these clarifications, the staff concludes that no further changes need to be made to the MRP-227, Rev. 1-A (ADAMS Accession No. ML19339G350) as endorsed in the NRC staff's letter dated February 20, 2020 (ADAMS ML20006D152).

Conditions and Plant-Specific Action Items

None.

ADAMS Accession Nos: Package: ML20141L313; Email: ML20141L314; SE: ML20141L315

Approval	Printed Name	Signature	Date
Technical Branch Chief	Hipolito Gonzalez	R/A via Email	5/26/2020
Projects Branch Chief	Dennis Morey	R/A via Email	5/28/2020

ML20141L315

5/28/2020

Kyle,

By letter dated December 3, 2019 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML19339G364), the Electric Power Research Institute (EPRI) submitted "Submittal of MRP-227 Revision 1-A, Final Published Version Electric Power Research Institute Topical Report, entitled 'Materials Reliability Program: Pressurized Water Reactor Internals Inspection and Evaluations Guideline,' for NRC [US Nuclear Regulatory Commission] Acceptance (CAC NO. MF7740)" (ADAMS Accession No. ML19339G348).

The NRC staff provided a draft safety evaluation (SE) to EPRI for comment via email dated May 14, 2020 (ADAMS Accession No. ML201227H669). By letter dated May 22, 2020 (ADAMS Accession No. ML20146A002), EPRI informed the NRC staff that there was no proprietary information in the draft SE and had no comments on its accuracy or clarity.

The NRC staff has found that the supplemental information provided, is acceptable for referencing in licensing applications for nuclear power plants to the extent specified and under the limitations delineated in the submittal and in the enclosed final SE. The final SE defines the basis for our acceptance of the supplemental information. A copy of the final SE is attached to this email.

Our acceptance applies only to material provided in the supplemental information. We do not intend to repeat our review of the acceptable material described in the supplemental information. When the supplemental information appears as a reference in license applications, our review will ensure that the material presented applies to the specific plant involved. License amendment requests that deviate from the supplemental information will be subject to a plant specific review in accordance with applicable review standards.

In accordance with the guidance provided on the NRC website, we request that EPRI publish an accepted version of the MRP-227, Rev 1-A within three months of receipt of this email. The accepted versions shall incorporate the final SE after the title page. Once the -A version of MRP-227, Rev 1-A has been verified by the NRC staff, licensees may then reference the supplemental information.

If you have any questions or require any additional information, please feel free to contact me at (301) 415 7297 or joseph.holonich@nrc.gov.

This email and the attached SE have been placed in ADAMS and made public.

Joe Holonich, Senior Project Manager
Office of Nuclear Reactor Regulation

ML20141L314

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MRP Materials Reliability Program ________________________ MRP 2020-013

May 22, 2020 EPRI Docket No. 99902021

U.S. Nuclear Regulatory Commission
Document Control Desk
11555 Rockville Pike
1 White Flint N; Mail Stop: 0-12-D2
ROCKVILLE, MD 20852

SUBJECT: Industry response to draft safety evaluation of EPRI report MRP-227-Revision 1-A

Dear Sir:

EPRI is in receipt of the NRC's draft safety evaluation (SE) report transmitted by email on 5/14/2020 (ML20127H293). The NRC staff requested that EPRI review this draft SE for accuracy, clarity, and proprietary information. The PWR utility industry representatives have confirmed that the draft SE contains no proprietary information. Furthermore, the industry representatives have no comments to this draft SE.

EPRI appreciates the continued dialogue with NRC staff associated with materials aging management.

If there are any concerns or questions, please contact the undersigned.

Sincerely,

Chris Koehler, Xcel Energy
MRP Research Integration Committee Chair

Brian Burgos, Sr. Program Manager
EPRI-MRP

Cc: MRP RIC Members

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MRP Materials Reliability Program MRP 2020-012

May 4, 2020 EPRI Docket No. 99902021

U.S. Nuclear Regulatory Commission
Document Control Desk
11555 Rockville Pike
1 White Flint N; Mail Stop: 0-12-D2
ROCKVILLE, MD 20852

Dear Sir:

EPRI is in receipt of NRC's letter dated 2/19/2020 (ML20006D152). This letter provides responses to the NRC's comments in the attachment. Please note that we believe NRC's comment/issue #4 has been previously addressed in NRC's Safety Evaluation Report for EPRI Topical Report MRP-227, Revision, "Materials Reliability Program: Pressurized Water Reactor Internals Inspection and Evaluations Guideline" (CAC NO.MF7223), dated 4/25/2019 (ML19081A001), specifically in section 3.6.3.

If you have questions, please advise the undersigned.

Sincerely,

Mike Hoehn II, Ameren-Missouri
MRP Research Integration Committee Chair

Brian Burgos, Sr. Program Manager
EPRI-MRP

Enclosure:
Responses to Nuclear Regulatory Commission Staff Comments on MRP-227, Revision 1-A

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Responses to Nuclear Regulatory Commission Staff Comments on MRP-227, Revision 1-A

Prepared for EPRI by:

Westinghouse Electric Company LLC
Cranberry, PA

Ref. Westinghouse letter LTR-ALMR-20-37

Attachment 1

Responses to Nuclear Regulatory Commission Staff Comments on MRP-227, Revision 1-A

1 Background and Purpose

MRP-227, Revision 1 was published in October 2015 [1] and transmitted to the U.S. Nuclear Regulatory Commission (NRC) for review and approval in December 2015 [2] with an errata letter issued in January 2017 to address several corrections [3]. The NRC staff conducted its review of MRP-227, Revision 1 and provided a safety evaluation in April 2019 [4].

The industry, through the Materials Reliability Program (MRP), updated MRP-227, Revision 1 to Revision 1-A, incorporating the various changes committed to during the response to NRC requests for additional information during the safety evaluation process. MRP-227, Revision 1-A was published in December 2019 [5] and sent to the NRC for final review and acceptance in letter MRP 2019-033 [6].

The NRC staff provided acceptance of MRP-227, Revision 1-A in a letter dated February 19, 2020 [7], stating:

> *The -A version of the TR incorporates the NRC staff's safety evaluation. Based on its review of MRP-227, Revision 1-A, the NRC staff has determined that the TR is acceptable to the extent delineated in the NRC staff's safety evaluation dated April 25, 2019 (ADAMS Accession No. ML19081A001).*

However, this was qualified by the inclusion of an enclosure table containing five (5) comments from the NRC staff on the published MRP-227, Revision 1-A report and the following statement in the letter:

> *Any facility referencing MRP-227, Revision 1-A should address the items contained in the enclosure to this letter.*

The purpose of this letter is to address these five comments generically. A utility referencing MRP-227, Revision 1-A may reference this letter to meet the recommendation of the NRC verification letter for MRP-227, Revision 1-A [7].

2 Industry Responses to NRC Staff Comments on MRP-227, Revision 1-A

The NRC staff comments on MRP-227, Revision 1-A from the enclosure of the verification letter [7] and the industry responses are included in Table 1.

Table 1: Responses to U.S. NRC Staff Comments on MRP-227, Revision 1-A as Provided in [7]

Issue No.	**Issue Topic Or Confirmation Topic**	**Issue Topic Gap in MRP-227, Rev. 1-A, or TR Topic Needing Confirmation from EPRI with Regard to Contents in Rev. 1-A**	**Topic Issue Needing Confirmation or Consideration from the EPRI MRP**	**Industry Response**
Content Issues in MRP-227, Rev. 1-A (the TR) Needing Attention and Potential Resolution by EPRI				
1	Combustion Engineering-design (CE-design) core support column bolts Item C1.1 in TR Table 4-5 "Expansion" Category Components	The line item entry for the core support column bolts in TR Table 3-2 (on page 3-24) identifies that the bolts are located in the lower support structure. In comparison to the line item entry for the bolts in TR Table 3-2, Item C1.1 in TR Table 4-5 (on page 4-30) identifies that the core support column bolts are located in core shroud assembly for CE-design plants with bolted core shroud assembly designs.	Either the line item entry for the core support column bolts in TR Table 3-2 needs to be moved from the lower support structure part of the table to the core shroud part of the table, or Line Item C1.1 in TR Table 4-5 needs to change the listed assembly from "**Core Shroud Assembly (Bolted)**" to "**Lower Support Structure**". TR Table 5-2 would need to be adjusted accordingly.	The core support column bolts are considered part of the CE lower support structure. Thus, the entry in Table 3-2 is correct in MRP-227, Revision 1-A. The Component assembly for C1.1 "Core support column bolts" in Table 4-5 should be "Lower Support Structure" instead of "Core Shroud Assembly (Bolted)". C1.1 also appears in Table 5-2 in the C1 "Core Shroud Assembly (Bolted) – Core shroud bolts" row; however, this entry does not include the component assembly for the C1.1 "Core support column bolts" and is correct in MRP-227, Revision 1-A. This incorrect component assembly for the CE Core Support Column Bolts in Table 4-5 was also present in MRP-227-A [8] and was not changed for MRP-227, Revision 1 or Revision 1-A. The core support column bolts are only applicable for one currently operating plant (Palisades). This one plant should make this component assembly name correction when implementing MRP-227, Revision 1-A. Additionally, this editorial change will be made in MRP-227, Revision 2.

Table 1: Responses to U.S. NRC Staff Comments on MRP-227, Revision 1-A as Provided in [7] (cont.)

Issue No.	**Issue Topic Or Confirmation Topic**	**Issue Topic Gap in MRP-227, Rev. 1-A, or TR Topic Needing Confirmation from EPRI with Regard to Contents in Rev. 1-A**	**Topic Issue Needing Confirmation or Consideration from the EPRI MRP**	**Industry Response**
2	CE-design core support barrel (CSB) assembly lower flange weld (LFW) Line item entry for the LFW in TR Table 3-2 (See line item on page 3-25 of the TR)	The line item for the CSB LFW in TR Table 3-2 (on page 3-25) identifies that the final inspection category for the LFW is "Primary". In comparison, TR Table 4-2 (the TR's examination table for Primary Components) does not include any line item entry for the CSB LFW. Instead, TR Table 4-2 includes Line Item C7 (on page 4-19) for what is defined as a flexure weld in the CSB and which is identified as a lead "Primary" category component for CE-designed plants.	If the CSB LFW is the same as the CSB flexure weld, then the line item entries for the weld in TR Table 3-2 and 4-2 need to be adjusted to make it clear that the CSB LFW and CSB flexure weld are the same component. Otherwise, if the CSB LFW is not the same component as the CSB flexure weld, then a line item entry for the CSB flexure weld needs to be added to TR Table 3-2 and a line item for the CSB LFW needs to be added to TR Table 4-2. TR Table 5-2 would need to be adjusted accordingly.	Item C7 "CSB Flexure Weld" in Table 4-2 of MRP-227, Revision 1-A is a more detailed name for the "Lower Core Barrel Flange Weld" in Table 3-2. The CSB flexure weld was named "Lower flange weld" in MRP-227-A [8], but this created potential confusion, since there are two welds located at the CE core support barrel lower flange. The fatigue concern is for the flexure weld (Primary Item C7), while the weld that fastens the lower flange to the barrel was included in Revision 1-A as C5.1 "Lower Girth Weld (LGW)". Section 3 of MRP-227 (all revisions) is provided as a background section summarizing the development of the reactor vessel internals aging management requirements in Sections 4 and 5. Per MRP-227, Revision 1-A, Section 7.3, the NEI 03-08 "Needed" requirement tables are Tables 4-1 through 4-9 and Tables 5-1 through 5-3. The background information in Section 3 is not included as an NEI 03-08 requirement. Section 3 provides valuable historic information from documents such as MRP-191 and MRP-232, but the requirements in Sections 4 and 5 are final and may be refined from the background in Section 3. Note that this potential for differences between the intermediate stage tables provided in Section 3 and the actual guidance tables provided in Sections 4 and 5 was addressed in the last paragraph of Section 3.3.2, just before the tables: *"Note that the component nomenclature used in theses tables is consistent with the bases documents and may not be the same as that used in the tables of Sections 4 and 5. Also, the final grouping (P, E, N, X) listed here is an intermediate result and does not represent the final updated inspection strategy as defined for this revision of the guidelines."*

Table 1: Responses to U.S. NRC Staff Comments on MRP-227, Revision 1-A as Provided in [7] (cont.)

Issue No.	**Issue Topic Or Confirmation Topic**	**Issue Topic Gap in MRP-227, Rev. 1-A, or TR Topic Needing Confirmation from EPRI with Regard to Contents in Rev. 1-A**	**Topic Issue Needing Confirmation or Consideration from the EPRI MRP**	**Industry Response**
3	CE-design core support barrel (CSB) assembly lower girth welds (LGWs) Item C5.1 in TR Table 4-5 "Expansion" category components	The line item entry for CSB LGWs in TR Table 3-2 (Page 3-25) identifies that the LGWs are "Primary" category components for CE-design plants. This line item includes a reference to Table Note 4 on the SCC and IE mechanisms and Final Group entries for the line item, with Note 4 stating "Upgraded to Primary" from "Expansion" in accordance with the NRC SER on MRP-227, Revision 0." The line item also includes a reference to Table Note 7 for the listed IASCC entry in the line item, with Note 7 stating: "Mechanism "IASCC" upgraded from "No Additional Measures" to "Primary for lower cylinder welds in accordance with NRC SER on MRP-227, Revision 0." In comparison to the line item entry for the CSB LGWs in TR Table 3-2, Item C5.1 in TR Table 4-5 (on Page 4-29) identifies that the CSB LGWs are "Expansion" category components for CE-design plants.	If the EPRI MRP downgraded the CE-design CSB LGWs to "Expansion" category components in response to one of our RAIs for the MRP-227, Revision 1 methodology and we accepted the RAI basis change in our April 25, 2019 safety evaluation for the MRP-227, Rev. 1 methodology, then the SCC, IASCC, IE and Final Group column references for the CSB LGWs in the Table 3-2 of the MRP-227, Rev. 1-A report will need to be fixed from "P" to "E", and Notes 4 and 7 for TR Table 3-2 will need to be adjusted accordingly. However, the existing TR Table 3-2 notes in the draft MRP-227, Rev. 1-A report appear to indicate that the CSB LGWs remain as "Primary" category components. If instead, the CSB LGWs remain as "Primary" category components for the program, then Item C5.1 for the CSB LGWs is incorrect and needs to be deleted from the scope of TR Table 4-5, and a new Item for the CSB LGWs would need to be developed and included in Table 4-2 of the MRP-227, Rev. 1-A report. Table 5-2 would need to be adjusted accordingly.	The "Lower Cylinder Girth Welds" in Table 3-2 of MRP-227, Rev. 1-A do not represent C5.1 "CSB Lower Girth Weld (LGW)" in Table 4-5. A review of the aging degradation mechanisms cited for each one makes this clear: the Table 3-2 lower cylinder girth welds were included for stress corrosion cracking (SCC), irradiation-assisted SCC, and irradiation embrittlement while the Table 4-5 CSB lower girth weld was included in the Expansion table for SCC and fatigue. Instead, the "Lower Cylinder Girth Welds" in Table 3-2 represent C6 "Middle Girth Weld (MGW)" in Table 4-2 of MRP-227, Rev. 1-A, which is a primary item, consistent with the Table 3-2 entry. Item C5.1 "CSB Lower Girth Weld (LGW)" is represented by the "Lower Core Barrel Flange" in Table 3-2. The aging degradation concern for the lower core barrel flange listed in Table 3-2 is the weld that connects the flange to the barrel cylinder, known as the "Lower Girth Weld". The degradation mechanisms and suggested inspection category listed in Table 3-2 for the lower core barrel flange match those for C5.1 in Table 4-5 of MRP-227, Rev. 1-A. As noted for Issue No. 2, Section 3 of MRP-227 (all revisions) is provided as a background section summarizing the development of the reactor vessel internals aging management requirements in Sections 4 and 5 and only Tables 4-1 through 4-9 and 5-1 through 5-3 are designated as NEI 03-08 "Needed" requirements per MRP-227, Rev. 1-A, Section 7.3. The background information in Section 3 is not included as an NEI 03-08 requirement. Section 3 provides valuable historic information from documents such as MRP-191 and MRP-232, but the requirements in Sections 4 and 5 are final and may be refined from the background in Section 3. Also, as noted for Issue No. 2, the last paragraph of Section 3.3.2 clarifies the potential for differences between the intermediate tables in Section 3 and the actual guidance tables in Sections 4 and 5.

Table 1: Responses to U.S. NRC Staff Comments on MRP-227, Revision 1-A as Provided in [7] (cont.)

Issue No.	**Issue Topic Or Confirmation Topic**	**Issue Topic Gap in MRP-227, Rev. 1-A, or TR Topic Needing Confirmation from EPRI with Regard to Contents in Rev. 1-A**	**Topic Issue Needing Confirmation or Consideration from the EPRI MRP**	**Industry Response**
4	CE-Design Incore Instrumentation (ICI) Thimble Tubes Line item entry for the "ICI Thimble Tubes-Lower" in TR Table 3-2 (See line item on page 3-26 of the TR)	The line item for the ICI thimble tubes in Table 3-2 (on page 3-26) of the MRP-227, Rev. 1-A report indicates that CE-design "ICI thimble tubes – lower" are "Existing Program" components for CE-designed plants; yet Table 4-8 in the report does not include any line item entry for "ICI Thimble-Tubes – Lower."	The EPRI MRP needs to establish whether the MRP-227, Rev. 1-A guidelines are continuing to define the "ICI Thimble Tubes-Lower" as "Existing Program" components for CE-design RVI aging management programs, and if so, whether the inclusion of the tubes in the "Existing Program" category is based on the designation of the ICI thimble tubes as ASME Section XI components or known site-specific programs for the tubes, therefore may need a revision to Table 4-8 to include a line item for the thimble tubes.	This comment has been resolved by the NRC staff SE on MRP-227, Revision 1-A, specifically in 3.6.3. Selected text is included here: *A/LAI 3 was included in Ref. 16 because MRP-227-A did not provide adequate guidance for applicants/licensees to document the details of the plant-specific existing programs in plant-specific RVI programs. The NRC staff notes that MRP-227-A did state, with regard to existing plant-specific programs for CE-design RVI. that "the guidance for in-core instrumentation (ICI) thimble tubes and thermal shield positioning pins is limited to plant-specific recommendations and thus have no generic reference, nor are they included in Table 4-8. The owner should review their specific design, upgrade status, and plant commitments for CE ICI thimble tubes."* And later in Section 3.6.3: *In MRP-227, Revision 1, the guidance, with regard to CE existing plant-specific items, has been eliminated. The NRC staff reviewed the status of the resolution of A/LAI 3 for the CE units implementing MRP-227-A. The staff has received submittals for 8 of the 10 operating CE units for review of the plant-specific RVI programs. The NRC staff has approved the RVI programs for all 8 of these units. A/LAI 3 was either not applicable or successfully resolved for all 8 of these units. For the two remaining CE units, the licensee does not have a commitment to submit the RVI program for NRC staff review.* *Therefore, A/LAI 3 is resolved or expected to be resolved as part of implementation of RVI programs based on MRP-227-A for all CE units that have commitments to submit RVI programs for staff review and approval. A/LAI 3 can be eliminated for CE-design RVI in the accepted version MRP-227, Revision 1.* The resolution of A/LAI 3 shows that while the "Existing" designation in Table 3-2 of MRP-227, Revision 1-A for the ICI Thimble Tubes-Lower is still correct, there is no need to add the component to Table 4-8.

Table 1: Responses to U.S. NRC Staff Comments on MRP-227, Revision 1-A as Provided in [7] (cont.)

Issue No.	**Issue Topic Or Confirmation Topic**	**Issue Topic Gap in MRP-227, Rev. 1-A, or TR Topic Needing Confirmation from EPRI with Regard to Contents in Rev. 1-A**	**Topic Issue Needing Confirmation or Consideration from the EPRI MRP**	**Industry Response**
Confirmations Needed from the EPRI MRP on Renumbering of Restructuring of Table Notes in MRP-227, Rev. 1-A				
5a	Confirmation of Changes to Tabular Footnote References for TR Tables 4-1 through 4-6 and 4-8 and 4-9	**<u>Notes for Item W6 in TR Table 4-3</u>**. For Westinghouse-design "Primary" components associated with Item W6 (i.e., baffle-former bolts) in TR Table 4-3, the applicable note designations for the bolts went from Notes 3, 6, and 9 in Table 4-3 of the MRP-227, Rev. 1 report to Notes 3, 4, 7, and 9 in Table 4-3 of the draft MRP-227, Rev. 1-A report.	(Note 1)	**W6**: Component notes 3, 6, and 9 from Table 4-3 of MRP-227, Revision 1 for component W6 "Baffle-Former Bolts" appear as Notes 3, 4, and 7 in Table 4-3 of MRP-227, Revision 1-A. The note numbers changed from "6" to "4" and from "9" to "7" due to the administrative deletion of notes 4 and 5, because neither of those notes were actually referenced in Table 4-3 in MRP-227, Revision 1. Component notes 8 and 9 were added into MRP-227, Revision 1-A for component W6 in accordance with the response to NRC RAI 8 and the clarification of the response to RAI 8 (see Appendix D of MRP-227, Revision 1-A).
5b	Confirmation of Changes to Tabular Footnote References for TR Tables 4-1 through 4-6 and 4-8 and 4-9	**<u>Notes for Item W1 in TR Table 4-3</u>**. For Westinghouse-design "Primary" components associated with Item W6 (i.e., CRGT guide cards) in TR Table 4-3, applicable note designation went from Note 7 in Table 4-3 of the MRP-227, Rev. 1 report to Note 5 in Table 4-3 of the draft MRP-227, Rev. 1-A report.	(Note 1)	**W1**: Component note 7 from Table 4-3 of MRP-227, Revision 1 for component W1 "CRGT Guide plates (cards)" appears as Note 5 in MRP-227, Revision 1-A. The note number changed from "7" to "5" due to the administrative deletion of notes 4 and 5, because neither of those notes were actually referenced in Table 4-3 in MRP-227, Revision 1. The note text was updated in MRP-227, Revision 1-A in accordance with the response to NRC RAI 19 on MRP-227, Revision 1 and the clarifications made on NRC RAI 19 (see Appendix D of MRP-227, Revision 1-A).

Table 1: Responses to U.S. NRC Staff Comments on MRP-227, Revision 1-A as Provided in [7] (cont.)

Issue No.	**Issue Topic Or Confirmation Topic**	**Issue Topic Gap in MRP-227, Rev. 1-A, or TR Topic Needing Confirmation from EPRI with Regard to Contents in Rev. 1-A**	**Topic Issue Needing Confirmation or Consideration from the EPRI MRP**	**Industry Response**
5c	Confirmation of Changes to Tabular Footnote References for TR Tables 4-1 through 4-6 and 4-8 and 4-9	**Notes for Items C5 and C6 in TR Table 4-2**. For CE-design "Primary" components associated with Item C5 (core support barrel upper flange weld) and Item C6 (i.e., core support barrel middle girth weld) in TR Table 4-2, the applicable note designations went from Notes 5 and 6 in Table 4-2 of the MRP-227, Rev. 1 report to Note 5 in Table 4-3 of the draft MRP-227, Rev. 1-A report. For some reason, the EPRI MPR did not reference the renumbered Note 5 in the "Examination Column entries for Items C5 and C6 in Table 4-2 of the MRP-227, Rev. 1-A report.	(Note 1) **Additional clarifications for notes associated with TR Table 4-2, Items C5 and C6**. For the change in the designation of Note 6 in Table 4-2 of TR MRP-227, Rev. 1 to Note 5 in Table 4-2 of MRP-227, Rev. 1-A, EPRI MRP to confirm whether or not the failure to reference the renumbered Note 5 in the "Examination Coverage" column entries for Item C5 and C6 in Table 4-2 of the MRP-227, Rev. 1-A was an inadvertent omission and whether or not the "Examination Coverage" column entries for Items C5 and C6 in Table 4-2 of the Rev. 1-A report should be amended to include reference to both renumbered Notes 4 and 5 (and not just renumbered Note 4).	**C5 and C6**: Component Notes 5 and 6 from Table 4-2 of MRP-227, Revision 1 for components C5 "Upper flange weld (UFW)" and C6 "Middle Girth Weld (MGW)" appear as Notes 4 and 5 in MRP-227, Revision 1-A. The note numbers changed from "5" and "6" to "4" and "5" due to the administrative deletion of Note 3 from Table 4-2 of Revision 1-A, because that note only referred to the core support columns (formerly C8 in Revision 1). The core support columns were moved to Table 4-5 (C6.3 in Revision 1-A) in accordance with the response to NRC RAI 9 on MRP-227, Revision 1 (see Appendix D of MRP-227, Revision 1-A). Component Note 6 from Table 4-2 of MRP-227, Revision 1 (now Note 5 in Revision 1-A) was removed from components C5 and C6 consistent with the notes included for these components in response to NRC RAI 29. Note 6 from MRP-227, Revision 1 required expansion to the uninspected portion of the C5 and C6 welds during the same outage if significant flaws were found in the Primary inspection sample. The Primary Component coverage for C5 and C6 in Table 4-2 of MRP-227, Revision 1-A is 100% of the accessible weld length of one side of the weld instead of the 25% required in MRP-227, Revision 1. The change in coverage made the MRP-227, Revision 1 Note 6 obsolete and unnecessary for C5 and C6. (Note that Revision 1 Note 6 was kept in Revision 1-A as the renumbered Note 5 because it was still referenced for C12 "Lower Support Structure Deep beams").

Table 1: Responses to U.S. NRC Staff Comments on MRP-227, Revision 1-A as Provided in [7] (cont.)				
Issue No.	**Issue Topic Or Confirmation Topic**	**Issue Topic Gap in MRP-227, Rev. 1-A, or TR Topic Needing Confirmation from EPRI with Regard to Contents in Rev. 1-A**	**Topic Issue Needing Confirmation or Consideration from the EPRI MRP**	**Industry Response**
5d	Confirmation of Changes to Tabular Footnote References for TR Tables 4-1 through 4-6 and 4-8 and 4-9	**Notes for Items W3 and W4 in TR Table 4-3**. For "Primary" components associated with Item W3 (core barrel upper flange weld) and Item W4 (i.e., core barrel lower girth weld) in TR Table 4-3, the applicable note designations went from Notes 8 and 10 in Table 4-3 of the MRP-227, Rev. 1 report to Note 6 in Table 4-3 of the draft MRP-227, Rev. 1-A report. The applicable note on minimum inspection coverages applying to the "Primary" core barrel welds in Items W3 and W4 of Table 4-3 in MRP-227, Rev. 1, was listed as Note 8. In Table 4-3 of MRP-227, Rev. 1-A, the note on minimum inspection coverages applying to Items W3 and W4 was renumbered to Note 6. For some reason this note did not carry forward as an applicable note for Table 4-3 of the report similar to manner that the analogous note for CE "Primary" core support barrel welds had carried over in Table 4-2 of the report. Thus, the "Examination Coverage' column entries for Items W3 and W4 in Table 4-3 of the MRP-227, Rev. 1-A report do not currently include reference to a note that is analogous to Note 10 in Table 4-3 of the MRP-227, Rev. 1 report.	(Note 1) **Additional clarifications for notes associated with TR Table 4-3, Items W3 and W4**. EPRI to confirm whether or not the "Examination Coverage" column entries for Items W3 and W4 in Table 4-3 of the MRP-227, Rev. 1-A report should include reference to a renumbered note that is analogous to Note 10 in Table 4-3 of the MRP-227, Rev. 1 report.	**W3 and W4**: Component note 8 from Table 4-3 of MRP-227, Revision 1 for components W3 "Upper flange Weld (UFW)" and W4 "Lower girth weld (LGW)" appears as Note 6 in MRP-227, Revision 1-A. The note number changed from "8" to "6" due to the administrative deletion of notes 4 and 5, because neither of those notes were actually referenced in Table 4-3 in MRP-227, Revision 1. The examination coverage within this note was updated in MRP-227, Revision 1-A in accordance with the response to NRC RAI 29 (see Appendix D of MRP-227, Revision 1-A). Component note 10 from Table 4-3 of MRP-227, Revision 1 was removed from component W3 and W4 consistent with the notes included in response to NRC RAI 29. Note 10 from MRP-227, Revision 1 required expansion to the uninspected portion of the W3 and W4 welds during the same outage if significant flaws were found in the Primary inspection sample. The Primary Component coverage for W3 and W4 in Table 4-3 of MRP-227, Revision 1-A is 100% of the accessible weld length of one side of the weld (either side for W3 and the outer diameter for W4) instead of the 25% required in MRP-227, Revision 1. The change in coverage made Note 10 obsolete and unnecessary.

Table 1: Responses to U.S. NRC Staff Comments on MRP-227, Revision 1-A as Provided in [7] (cont.)

Issue No.	**Issue Topic Or Confirmation Topic**	**Issue Topic Gap in MRP-227, Rev. 1-A, or TR Topic Needing Confirmation from EPRI with Regard to Contents in Rev. 1-A**	**Topic Issue Needing Confirmation or Consideration from the EPRI MRP**	**Industry Response**
5e	Confirmation of Changes to Tabular Footnote References for TR Tables 4-1 through 4-6 and 4-8 and 4-9	**Notes for Items W4.2 and W4.3 in TR Table 4-6**. For "Expansion" components associated with Item W4.2 (core barrel middle axial welds) and Item W4.3 (i.e., core barrel lower axial welds) in TR Table 4-6, the applicable note designations were renumbered from Notes 2 and 3 in TR Table 4-6 of the MRP-227, Rev. 1 report to Notes 3 and 4 in TR Table 4-6 of the draft MRP-227, Rev. 1-A report.	(Note 1)	**W4.2 and W4.3**: Components W4.2 "Middle Axial Welds (MAW)" and W4.3 "Lower Axial Welds (LAW)" in Table 4-6 of MRP-227, Revision 1 included Note 2; however, the table did not include a Note 3, as indicated in [7]. Note 2 was incorrectly referenced for W4.2 and W4.3 in Table 4-6 of MRP-227, Revision 1 because those components already indicated in the "Expansion Coverage" column that the inspection should be conducted from the outer diameter of the barrel because the inner diameter of the barrel is inaccessible for these two welds. Component Notes 5 and 6 were added into MRP-227, Revision 1-A for components W4.2 and W4.3 in accordance with the response to NRC RAI 29 (see Appendix D of MRP-227, Revision 1-A).

Note:

1. Unless addressed in subsequent paragraphs of this column entry, the staff requests that EPRI MRP confirm that changes to the note number designation or the contents of notes on TR Section Table component items were based on either: (1) administrative reshuffling of the notes, or (2) adjustments of the note criteria based on the EPRI MRP's resolution of staff-issued requests for additional information (RAIs) on the MRP-227, Rev. 1 methodology, where the response bases were accepted by the staff.

3 Recommendations for Utility Use of this Letter

Utilities that reference MRP-227, Revision 1-A may reference the applicable portions of this letter to resolve the following statement in the NRC verification letter for MRP-227, Revision 1-A [7]:

> *Any facility referencing MRP-227, Revision 1-A should address the items contained in the enclosure to this letter.*

Most of these responses do not require changes to the reactor vessel internals aging management program of a particular facility and will not impact the contents of MRP-227, Revision 1-A. The only exception is for Item 1 in Table 1, which requires the correction of the component assembly naming of MRP-227, Revision 1-A Table 4-5 entry C1.1 "Core support column bolts" to be the "Lower Support Structure" instead of the "Core Shroud Assembly (Bolted)". Note that this only applies to one currently-operating plant.

The applicability of the comments and comment responses in Table 1 are provided in Table 2 based on plant design.

Table 2: Plant Design Applicability of Comments and Comment Resolutions in Table 1

Plant Design	Comment Applicability and Actions
Babcock & Wilcox Plants	None of the comments are applicable. No further action needed.
Westinghouse Plants	Items 5a, 5b, 5d, and 5e are applicable. The note numbering changes were confirmed to be administrative, editorial adjustments and do not require further action beyond referencing the explanation provided in Table 1.

CE Plants (all designs)	Items 1, 2, 3, 4, and 5c are applicable, depending on plant design:
	• Item 1 – Applicable to CE plants with bolted core shroud assemblies (one currently-operating unit)
	  ○ Requires an update to the component assembly naming for Expansion component C1.1 in Table 4-5 of MRP-227, Revision 1-A
	• Item 2 – Applicable to CE plants with a core support barrel lower flange flexure weld
	  ○ Addressed in the comment resolution and does not require further action beyond referencing the explanation provided in Table 1
	• Item 3 – Applicable to all CE plants
	  ○ Addressed in the comment resolution and does not require further action beyond referencing the explanation provided in Table 1
	• Item 4 – Applicable to CE plants with affected ICI thimble tubes-Lower
	  ○ Addressed in the comment resolution and does not require further action beyond referencing the explanation provided in Table 1
	• Item 5c – Applicable to all CE plants
	  ○ Addressed in the comment resolution and does not require further action beyond referencing the explanation provided in Table 1

4 References

1. *Materials Reliability Program: Pressurized Water Reactor Internals Inspection and Evaluations Guideline (MRP-227, Revision 1)*, EPRI, Palo Alto, CA: 2015, 3002005349.
2. Materials Reliability Program Letter, MRP 2015-040, "Report Transmittal: *Materials Reliability Program: Pressurized Water Reactor Internals Inspection and Evaluations Guideline (MRP-227, Revision 1)*, EPRI, Palo Alto, CA, 2015, 3002005349. Ref. EPRI Project Number 689," December 21, 2015. (ML15358A046)
3. Materials Reliability Program Letter, MRP 2017-004, "Transmittal of Corrections to EPRI Report 3002005349, Materials Reliability Program: Pressurized Water Reactor Internals Inspection and Evaluation Guidelines (MRP-227, Revision 1)," January 18, 2017. (ML17024A252)
4. U.S. Nuclear Regulatory Commission Safety Evaluation, "Final Safety Evaluation for Electric Power Research Institute Topical Report MRP-227, Revision 1, "Materials Reliability Program: Pressurized Water Reactor Internals Inspection and Evaluations Guideline," (CAC No. MF7223; EPID L-2016-TOP-0001)," April 25, 2019. (ML19081A001)
5. *Materials Reliability Program: Pressurized Water Reactor Internals Inspection and Evaluation Guidelines (MRP-227, Revision 1-A)*. EPRI, Palo Alto, CA: 2019. 3002017168.
6. Materials Reliability Program Letter, MRP 2019-033, "Submittal of MRP-227 Revision 1-A, Final Published Version Electric Power Research Institute Topical Report, entitled "Materials Reliability Program: Pressurized Water Reactor Internals Inspection and Evaluations Guideline," for US NRC Acceptance (CAC NO. MF7740)," December 3, 2019. (ML19339G364)
7. U.S Nuclear Regulatory Commission Letter, "U.S. Nuclear Regulatory Commission Verification Letter for Electric Power Research Institute Topical Report MRP-227, Revision 1, 'Materials Reliability Program: Pressurized Water Reactor Internals Inspection and Evaluations Guideline' (L-2019-TOP-0053)," February 19, 2020. (ML20006D152)
8. *Materials Reliability Program: Pressurized Water Reactor Internals Inspection and Evaluation Guidelines (MRP-227-A).* EPRI, Palo Alto, CA: 2011. 1022863.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

February 19, 2020

Brian Burgos
MRP Program Manager
Electric Power Research Institute
3420 Hillview Avenue
Palo Alto, CA 94304

SUBJECT: U.S. NUCLEAR REGULATORY COMMISSION VERIFICATION LETTER FOR ELECTRIC POWER RESEARCH INSTITUTE TOPICAL REPORT MRP 227, REVISION 1, "MATERIALS RELIABILITY PROGRAM: PRESSURIZED WATER REACTOR INTERNALS INSPECTION AND EVALUATIONS GUIDELINE" (L-2019-TOP-0053)

Dear Mr. Burgos:

By letter dated December 3, 2019 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML19339G364), the Electric Power Research Institute submitted for U.S. Nuclear Regulatory Commission (NRC) staff review topical report MRP 227, Revision 1-A, "Materials Reliability Program: Pressurized Water Reactor Internals Inspection and Evaluations Guideline."

The -A version of the TR incorporates the NRC staff's safety evaluation. Based on its review of MRP-227, Revision 1-A, the NRC staff has determined that the TR is acceptable to the extent delineated in the NRC staff's safety evaluation dated April 25, 2019 (ADAMS Accession No. ML19081A001). Any facility referencing MRP-227, Revision 1-A should address the items contained in the enclosure to this letter.

If you have any questions or require any additional information, please feel free to contact the Project Manager for this topical report, Joseph Holonich, at 301-415-7297 or joseph.holonich@nrc.gov.

Sincerely,

/RA/

Dennis C. Morey, Chief
Licensing Processes Branch
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Enclosure: As stated
Docket No. 99902021

U.S. Nuclear Regulatory Commission Staff Comments on
MRP 227, Revision 1-A, "Materials Reliability Program: Pressurized Water Reactor Internals Inspection and Evaluations Guideline."

Table 1 – Summary of MRP-227, Rev. 1-A Issues or Contents Needing EPRI Confirmation

Issue No.	Issue Topic Or Confirmation Topic	Issue Topic Gap in MRP-227, Rev. 1-A, or TR Topic Needing Confirmation from EPRI with Regard to Contents in Rev. 1-A	Topic Issue Needing Confirmation or Consideration from the EPRI MRP
Content Issues in MRP-227, Rev. 1-A (the TR) Needing Attention and Potential Resolution by EPRI			
1	Combustion Engineering-design (CE-design) core support column bolts Item C1.1 in TR Table 4-5) "Expansion" Category Components	The line item entry for the core support column bolts in TR Table 3-2 (on page 3-24) identifies that the bolts are located in the lower support structure. In comparison to the line item entry for the bolts in TR Table 3-2, Item C1.1 in TR Table 4-5 (on page 4-30) identifies that the core support column bolts are located in core shroud assembly for CE-design plants with bolted core shroud assembly designs.	Either the line item entry for the core support column bolts in TR Table 3-2 needs to be moved from the lower support structure part of the table to the core shroud part of the table, or Line Item C1.1 in TR Table 4-5 needs to change the listed assembly from "**Core Shroud Assembly (Bolted)**" to "**Lower Support Structure**". TR Table 5-2 would need to be adjusted accordingly.
2	CE-design core support barrel (CSB) assembly lower flange weld (LFW) Line item entry for the LFW in TR Table 3-2 (See line item on page 3-25 of the TR)	The line item for the CSB LFW in TR Table 3-2 (on page 3-25) identifies that the final inspection category for the LFW is "Primary". In comparison, TR Table 4-2 (the TR's examination table for Primary Components) does not include any line item entry for the CSB LFW. Instead, TR Table 4-2 includes Line Item C7 (on page 4-19) for what is defined as a flexure weld in the CSB and which is identified as a lead "Primary" category component for CE-designed plants.	If the CSB LFW is the same as the CSB flexure weld, then the line item entries for the weld in TR Table 3-2 and 4-2 need to be adjusted to make it clear that the CSB LFW and CSB flexure weld are the same component. Otherwise, if the CSB LFW is not the same component as the CSB flexure weld, then a line item entry for the CSB flexure weld needs to be added to TR Table 3-2 and a line item for the CSB LFW needs to be added to TR Table 4-2. TR Table 5-2 would need to be adjusted accordingly.

Issue No.	Issue Topic Or Confirmation Topic	Issue Topic Gap in MRP-227, Rev. 1-A, or TR Topic Needing Confirmation from EPRI with Regard to Contents in Rev. 1-A	Topic Issue Needing Confirmation or Consideration from the EPRI MRP
3	CE-design core support barrel (CSB) assembly lower girth welds (LGWs) Item C5.1 in TR Table 4-5 “Expansion” category components	The line item entry for CSB LGWs in TR Table 3-2 (Page 3-25) identifies that the LGWs are “Primary” category components for CE-design plants. This line item includes a reference to Table Note 4 on the SCC and IE mechanisms and Final Group entries for the line item, with Note 4 stating “Upgraded to Primary” from “Expansion” in accordance with the NRC SER on MRP-227, Revision 0.” The line item also includes a reference to Table Note 7 for the listed IASCC entry in the line item, with Note 7 stating: “Mechanism “IASCC” upgraded from “No Additional Measures” to “Primary for lower cylinder welds in accordance with NRC SER on MRP-227, Revision 0.” In comparison to the line item entry for the CSB LGWs in TR Table 3-2, Item C5.1 in TR Table 4-5 (on Page 4-29) identifies that the CSB LGWs are “Expansion” category components for CE-design plants.	If the EPRI MRP downgraded the CE-design CSB LGWs to “Expansion” category components in response to one of our RAIs for the MRP-227, Revision 1 methodology and we accepted the RAI basis change in our April 25, 2019 safety evaluation for the MRP-227, Rev. 1 methodology, then the SCC, IASCC, IE and Final Group column references for the CSB LGWs in the Table 3-2 of the MRP-227, Rev. 1-A report will need to be fixed from “P” to “E”, and Notes 4 and 7 for TR Table 3-2 will need to be adjusted accordingly. However, the existing TR Table 3-2 notes in the draft MRP-227, Rev. 1-A report appear to indicate that the CSB LGWs remain as “Primary” category components. If instead, the CSB LGWs remain as “Primary” category components for the program, then Item C5.1 for the CSB LGWs is incorrect and needs to be deleted from the scope of TR Table 4-5, and a new Item for the CSB LGWs would need to be developed and included in Table 4-2 of the MRP-227, Rev. 1-A report. Table 5-2 would need to be adjusted accordingly.
4	CE-Design Incore Instrumentation (ICI) Thimble Tubes Line item entry for the “ICI Thimble Tubes-Lower” in TR Table 3-2 (See line item on page 3-26 of the TR)	The line item for the ICI thimble tubes in Table 3-2 (on page 3-26) of the MRP-227, Rev. 1-A report indicates that CE-design “ICI thimble tubes – lower” are “Existing Program” components for CE-designed plants; yet Table 4-8 in the report does not include any line item entry for “ICI Thimble-Tubes – Lower.”	The EPRI MRP needs to establish whether the MRP-227, Rev. 1-A guidelines are continuing to define the “ICI Thimble Tubes-Lower” as “Existing Program” components for CE-design RVI aging management programs, and if so, whether the inclusion of the tubes in the “Existing Program” category is based on the designation of the ICI thimble tubes as ASME Section XI components or known site-specific programs for the tubes, therefore may need a revision to Table 4-8 to include a line item for the thimble tubes.

Issue No.	Issue Topic Or Confirmation Topic	Issue Topic Gap in MRP-227, Rev. 1-A, or TR Topic Needing Confirmation from EPRI with Regard to Contents in Rev. 1-A	Topic Issue Needing Confirmation or Consideration from the EPRI MRP
Confirmations Needed from the EPRI MRP on Renumbering of Restructuring of Table Notes in MRP-227, Rev. 1-A			
5	Confirmation of Changes to Tabular Footnote References for TR Tables 4-1 through 4-6 and 4-8 and 4-9	<u>Notes for Item W6 in TR Table 4-3</u>. For Westinghouse-design "Primary" components associated with Item W6 (i.e., baffle-former bolts) in TR Table 4-3, the applicable note designations for the bolts went from Notes 3, 6, and 9 in Table 4-3 of the MRP-227, Rev. 1 report to Notes 3, 4, 7, and 9 in Table 4-3 of the draft MRP-227, Rev. 1-A report.	Unless addressed in subsequent paragraphs of this column entry, the staff requests that EPRI MRP confirm that changes to the note number designation or the contents of notes on TR Section Table component items were based on either: (1) administrative reshuffling of the notes, or (2) adjustments of the note criteria based on the EPRI MRP's resolution of staff-issued requests for additional information (RAIs) on the MRP-227, Rev. 1 methodology, where the response bases were accepted by the staff. <u>Additional clarifications for notes associated with TR Table 4-2, Items C5 and C6</u>. For the change in the designation of Note 6 in Table 4-2 of TR MRP-227, Rev. 1 to Note 5 in Table 4-2 of MRP-227, Rev. 1-A, EPRI MRP to confirm whether or not the failure to reference the renumbered Note 5 in the "Examination Coverage" column entries for Item C5 and C6 in Table 4-2 of the MRP-227, Rev. 1-A was an inadvertent omission and whether or not the "Examination Coverage" column entries for Items C5 and C6 in Table 4-2 of the Rev. 1-A report should be amended to include reference to both renumbered Notes 4 and 5 (and not just renumbered Note 4). <u>Additional clarifications for notes associated with TR Table 4-3, Items W3 and W4</u>. EPRI to confirm whether or not the "Examination Coverage" column entries for Items W3 and W4 in Table 4-3 of the MRP-227, Rev. 1-A report should include reference to a renumbered note that is analogous to Note 10 in Table 4-3 of the MRP-227, Rev. 1 report.
		<u>Notes for Item W1 in TR Table 4-3</u>. For Westinghouse-design "Primary" components associated with Item W6 (i.e., CRGT guide cards) in TR Table 4-3, applicable note designation went from Note 7 in Table 4-3 of the MRP-227, Rev. 1 report to Note 5 in Table 4-3 of the draft MRP-227, Rev. 1-A report.	
		<u>Notes for Items C5 and C6 in TR Table 4-2.</u> For CE-design "Primary" components associated with Item C5 (core support barrel upper flange weld) and Item C6 (i.e., core support barrel middle girth weld) in TR Table 4-2, the applicable note designations went from Notes 5 and 6 in Table 4-2 of the MRP-227, Rev. 1 report to Note 5 in Table 4-3 of the draft MRP-227, Rev. 1-A report. For some reason, the EPRI MPR did not reference the renumbered Note 5 in the "Examination Column entries for Items C5 and C6 in Table 4-2 of the MRP-227, Rev. 1-A report.	
		<u>Notes for Items W3 and W4 in TR Table 4-3.</u> For "Primary" components associated with Item W3 (core barrel upper flange weld) and Item W4 (i.e., core barrel lower girth weld) in TR Table 4-3, the applicable note designations went from Notes 8 and 10 in Table 4-3 of the MRP-227, Rev. 1 report to Note 6 in Table 4-3 of the draft MRP-227, Rev. 1-A report. The applicable note on minimum inspection coverages applying to the "Primary" core barrel welds in Items W3 and W4 of Table 4-3 in MRP-227, Rev. 1, was listed as Note 8. In Table 4-3 of MRP-227, Rev. 1-A, the note on minimum inspection coverages applying to Items W3 and W4 was renumbered to Note 6. For some reason this note did not carry forward as an applicable note for Table 4-3 of the report similar to manner that the analogous note for CE "Primary" core support barrel welds had carried over in Table 4-2 of the report. Thus, the "Examination Coverage' column entries for Items W3 and W4 in Table 4-3 of the MRP-227, Rev. 1-A report do not currently include reference to a note that is analogous to Note 10 in Table 4-3 of the MRP-227, Rev. 1 report.	

Issue No.	Issue Topic Or Confirmation Topic	Issue Topic Gap in MRP-227, Rev. 1-A, or TR Topic Needing Confirmation from EPRI with Regard to Contents in Rev. 1-A	Topic Issue Needing Confirmation or Consideration from the EPRI MRP
		Notes for Items W4.2 and W4.3 in TR Table 4-6. For "Expansion" components associated with Item W4.2 (core barrel middle axial welds) and Item W4.3 (i.e., core barrel lower axial welds) in TR Table 4-6, the applicable note designations were renumbered from Notes 2 and 3 in TR Table 4-6 of the MRP-227, Rev. 1 report to Notes 3 and 4 in TR Table 4-6 of the draft MRP-227, Rev. 1-A report.	

SUBJECT: U.S. NUCLEAR REGULATORY COMMISSION VERIFICATION LETTER FOR ELECTRIC POWER RESEARCH INSTITUTE TOPICAL REPORT MRP 227, REVISION 1, "MATERIALS RELIABILITY PROGRAM: PRESSURIZED WATER REACTOR INTERNALS INSPECTION AND EVALUATIONS GUIDELINE" (L-2019-TOP-0053) DATED FEBRUARY 19, 2020

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April 25, 2019

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MRP Program Manager
Electric Power Research Institute
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Palo Alto, CA 94304

SUBJECT: FINAL SAFETY EVALUATION FOR ELECTRIC POWER RESEARCH INSTITUTE TOPICAL REPORT MRP-227, REVISION 1, "MATERIALS RELIABILITY PROGRAM: PRESSURIZED WATER REACTOR INTERNALS INSPECTION AND EVALUATIONS GUIDELINE" (CAC NO.MF7223; EPID L-2016-TOP-0001)

Dear Mr. Burgos:

By letter dated December 21, 2015 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML15358A046), the Electric Power Research Institute (EPRI) submitted for U.S. Nuclear Regulatory Commission (NRC) staff review topical report MRP 227, Revision 1, "Materials Reliability Program: Pressurized Water Reactor Internals Inspection And Evaluations Guideline." By letter dated February 28, 2019 the NRC staff issued its draft safety evaluation (SE) (ADAMS Accession No. ML18275A069).

By letter dated April 5, 2019 (ADAMS Accession No. ML19099A112), EPRI provided comments on the NRC draft SE. The comments provided by EPRI are contained in the attachment to the enclosed SE.

The NRC staff has found that MRP-227, Revision 1 is acceptable for referencing in licensing applications for nuclear power plants to the extent specified and under the limitations delineated in the TR and in the enclosed final SE. The final SE defines the basis for our acceptance of the TR.

Our acceptance applies only to material provided in the subject TR. We do not intend to repeat our review of the acceptable material described in the TR. When the TR appears as a reference in license applications, our review will ensure that the material presented applies to the specific plant involved. License amendment requests that deviate from this TR will be subject to a plant-specific review in accordance with applicable review standards.

In accordance with the guidance provided on the NRC website, we request that EPRI publish an accepted version of the TR within three months of receipt of this letter. The accepted-for-use version shall incorporate this letter and the enclosed final SE after the title page. Also, it must contain historical review information, including NRC requests for additional information (RAIs) and your responses. The accepted versions shall include a "-A" (designating accepted) following the TR identification symbol.

As an alternative to including the RAIs and RAI responses behind the title page, if changes to the TRs were provided to the NRC staff to support the resolution of RAI responses, and the NRC staff reviewed and accepted those changes as described in the RAI responses, there are two ways that the accepted version can capture the RAIs:

1. The RAIs and RAI responses can be included as an Appendix to the accepted version.
2. The RAIs and RAI responses can be captured in the form of a table (inserted after the final SE) which summarizes the changes as shown in the accepted version of the TR. The table should reference the specific RAIs and RAI responses which resulted in any changes, as shown in the accepted version of the TR.

If future changes to the NRC's regulatory requirements affect the acceptability of this TR, EPRI will be expected to revise the TR appropriately. Licensees referencing this TR would be expected to justify its continued applicability or evaluate their plant using the revised TR.

If you have any questions or require any additional information, please feel free to contact the NRC Project Manager for the review, Joseph Holonich at (301) 415-7297 or joseph.holonich@nrc.gov.

Sincerely,

Dennis C. Morey, Chief
Licensing Processes Branch
Division of Licensing Projects
Office of Nuclear Reactor Regulation

Docket No. 99902021

Enclosure:
Final SE

FINAL SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

"MATERIALS RELIABILITY PROGRAM: PRESSURIZED WATER REACTOR INTERNALS INSPECTION AND EVALUATION GUIDELINE (MRP-227 REVISION 1)"

1.0 INTRODUCTION

By letter dated December 21, 2015, the Electric Power Research Institute (EPRI) Materials Reliability Program (MRP) submitted "Materials Reliability Program: Pressurized Water Reactor Internals Inspection and Evaluation Guidelines (MRP-227, Revision 1)" (Ref. 1). EPRI stated that the report was being transmitted as a means of exchanging information with the Nuclear Regulatory Commission (NRC) staff for the purpose of supporting generic regulatory improvements related to the methodologies for verifying pressurized water reactor (PWR) internals integrity throughout the life of the plant, including the extended operating period authorized by license renewal in accordance with Title 10 of the *Code of Federal Regulation* (10 CFR) Part 54, "Requirements For Renewal of Operating Licenses for Nuclear Power Plants." On January 18, 2017, EPRI submitted a letter containing errata to the report (Ref. 2). By letters dated October 16, 2017, January 30, 2018, and September 28, 2018, EPRI provided responses to the NRC staff requests for additional information (RAIs) (Refs. 3, 4, and 5). EPRI provided additional supplemental information via letter dated May 17, 2018 (Ref. 6).

EPRI requested that the NRC staff issue a safety evaluation (SE) on MRP-227, Revision 1. EPRI further requested that the NRC staff review the incremental changes made to MRP-227-A, many of which were requested by NRC staff, rather than a complete re-review of the entire document. The incremental changes are detailed in Appendix C of the MRP-227, Revision 1 document. By letter dated October 5, 2017 (Ref. 7), EPRI provided several EPRI technical reports (Ref. 8-14) for information only to support the NRC staff review.

2.0 REGULATORY EVALUATION

Part 54 of 10 CFR addresses the requirements for plant license renewal. The regulation at 10 CFR 54.21, "Administrative review of applications; hearings," requires that each application for license-renewal (LR) contain an integrated plant assessment (IPA) and an evaluation of time limited aging analyses (TLAAs). The IPA shall identify and list those structures and components subject to an aging management review (AMR) and demonstrate that the effects of aging (i.e., cracking, loss of material, loss of fracture toughness, dimensional changes, loss of preload) will be adequately managed so that their intended functions will be maintained consistent with the current licensing basis (CLB) for the period of extended operation as required by 10 CFR 54.29, "Standards of Issuance of a Renewed License." In addition, 10 CFR 54.22, "Contents of Application--Technical Specifications," requires that a LR application include any technical specification (TS) changes or additions necessary to manage the effects of aging during the period of extended operation as part of the LR application.

Structures and components subject to an AMR shall encompass those structures and components that (1) perform an intended function, as described in 10 CFR 54.4, "Written communications," without moving parts or without a change in configuration or properties and (2) are not subject to replacement based on a qualified life or specified time period. These structures and components are referred to as "passive" and "long-lived" structures and components, respectively. The scope of components considered for inspection under MRP-227, Revision 1 guidance includes core support structures (typically denoted as

Examination Category B-N-3 by the American Society of Mechanical Engineers (ASME) Code, Section XI) and those reactor vessel internal (RVI) components that serve an intended LR safety function pursuant to criteria in 10 CFR 54.4(a)(1). The scope of the program does not include active RVI components (e.g., vent valve discs, shafts, or hinge pins), or consumable items such as fuel assemblies, reactivity control assemblies, and nuclear instrumentation. These components are not within the scope of the components that are required to be subject to an Aging Management Program (AMP), as defined by the criteria set in 10 CFR 54.21(a)(1).

Some owners of PWR units were granted renewed licenses contingent on a commitment to conform to the recommendations specified in NUREG-1801, "Generic Aging Lessons Learned (GALL) Report," Revision 1, AMP XI.M16, "PWR Vessel Internals." AMP XI.M16 requires that the applicant provide a commitment in the Final Safety Analysis Review (FSAR) supplement to (a) participate in the industry programs for investigating and managing aging effects on RVI components; (b) evaluate and implement the results of the industry programs as applicable to the RVI components; and (c) upon completion of these programs, but not less than 24 months before entering the period of extended operation, submit an inspection plan for RVI components to the NRC for review.

Each applicant/licensee that made a commitment to conform to the recommendation specified in NUREG-1801, Revision 1, AMP XI.M16 also made a commitment in its FSAR that it will implement the industry-developed AMP for its RVI components.

On January 12, 2009, EPRI submitted for NRC staff review and approval the MRP Report 1016596 (MRP-227), Revision 0, "PWR Internals Inspection and Evaluation Guidelines" (Ref. 15), which was intended as guidance for the use of applicants in developing their plant-specific AMP for RVI components.

Subsequent to the submittal of MRP-227 and prior to the issuance of the SE on MRP-227, Revision 0 (Ref. 16), NUREG-1801, Revision 2 (the GALL Report, Revision 2) was issued, providing new AMR line items and aging management guidance in AMP XI.M16A. This GALL AMP was based on NRC staff expectations for the guidance to be provided in MRP-227-A (Ref. 17).

Revision 1 to the final SE regarding MRP-227, Revision 0, was issued on December 16, 2011 (Ref. 16), with seven conditions and eight applicant/licensee action items. The topical report (TR) conditions were specified to ensure that certain information was revised generically in the final NRC-approved version of MRP-227 (MPR-227-A). The applicant/licensee action items were specified for applicant/licensees to address plant-specific issues which could not be resolved generically in Revision 1 of the final SE on MRP-227-A. On January 9, 2012, EPRI published MRP-227-A (Ref. 17).

MRP-227-A contains a discussion of the technical basis for the development of plant-specific AMPs for RVI components in PWR vessels and also provides inspection and evaluation guidelines for PWR applicants to use in their plant-specific AMPs. MRP-227-A provides the basis for renewed license holders to develop plant-specific inspection plans to manage aging effects on RVI components, as described by their FSAR commitment.

Since the GALL Report, Revision 2 (Ref. 18) was published prior to the issuance of the final SE of MRP-227-A, the NRC staff published License Renewal Interim Staff Guidance LR-ISG-2011-04: "Updated Aging Management Criteria for Reactor Vessel Internal Components

for Pressurized Water Reactors" (Ref. 19) which modifies the guidance of AMP XI.M16A to be consistent with MRP-227-A.

Additional Guidance Used or Referenced in Subsequent License Renewal Applications

For subsequent license renewal applications (SLRAs), the NRC staff addressed aging management criteria for PWR RVI components in Section 3.1.2.2.9, "*Aging Management of Pressurized Water Reactor Vessel Internals (Applicable to Subsequent License Renewal Periods Only),"* of NUREG-2912, "Standard Review Plan for Review of Subsequent License Renewal Applications for Nuclear Power Plants" (i.e., the SRP-SLR [subsequent license renewal] report [Ref. 20]), and in the NRC staff update of GALL-SLR AMP XI.M16A as given in NUREG-2191, "Generic Aging Lesson Learned for Subsequent License Renewal (GALL-SLR) Report" (Volumes 1 and 2 [Ref. 21]).

The aging management guidelines of MRP-227-A were developed based on the assumption of a plant life of 60 calendar years (e.g., a period of extended operation (PEO) of 20 years beyond the initial license). Therefore, SRP-SLR Section 3.1.2.2.9 and GALL-SLR AMP XI.M16A indicate that MRP-227-A report may be used as the starting point for an RVI AMP that will be defined in a PWR SLRA, subject to performance of a gap analysis. The purpose of the gap analysis is to determine if the augmented inspection-and-evaluation (IE) criteria for applicable RVI components in the MRP-227-A report need to be adjusted to account for an 80-year service life. Criteria and minimum expectations for implementing PWR RVI gap analyses have been described in the staff's updated version of the GALL-SLR AMP, as given in NUREG-2191.

MRP-227, Revision 1 was also developed based on the assumption of 60 calendar years of operation. If the NRC staff finds MRP-227, Revision 1, to be an acceptable basis for an RVI AMP for the PEO, MRP-227, Revision 1, as modified by this SE, could also be used as a starting point for performing a gap analysis in order to develop an RVI AMP for 60-80 years subsequent PEO (SPEO). Use of MRP-227, Revision 1 as a starting point for the gap analysis would require a LR applicant to identify an exception to SLR-GALL AMP XI.M16A.

3.0 TECHNICAL EVALUATION

3.1 NRC Staff Evaluation – Changes from MRP-227-A to MRP-227, Revision. 1

3.1.1 Section 1 Changes

The following wording was added to the executive summary:

> These guidelines are not intended to reduce, alter, or otherwise affect current American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel (B&PV) Code Section XI or plant-specific licensing inservice (sic) inspection requirements. Where ASME Code Section XI examinations are credited for aging management as 'existing programs,' these guidelines provide the specificity considered necessary to ensure that the examinations meet the intent for which they are credited.

The NRC staff finds this change acceptable.

The TR endorses WCAP-17096-NP, Revision 2, "Reactor Internals Acceptance Criteria Methodology and Data Requirements" (Ref. 22), or the latest NRC-approved version of the report, as an acceptable method for performing engineering evaluations of examination results that do not meet the acceptance criteria in TR Section 1 page 1-3, Section 2.1 page 2-2, Section 5 page 5-1, Section 6 page 6-1, and Section 7.5 page 7-2. Section 7.5, "Implementation Requirements," states that "…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." Section 7.5 further states that 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 accepted evaluation methods (i.e., ASME Code Section XI, WCAP-17096-NP or equivalent method).

The staff finds the use of WCAP-17096-NP acceptable, since the implementation requirements for MRP-227, Revision 1 in TR Section 7.5 specify that NRC-approved evaluation methods shall be used, which ensures that the NRC-approved version of WCAP-17096-NP will be used. The current accepted-for-use version of the report is WCAP-17096-NP-A Revision 2 (Ref. 23), and the NRC staff acceptance letter is Reference 24.

In TR Section 2.4, clarifying statements were added regarding expectations for users of the I&E guidelines. This is further addressed in Section 3.5 of this SE, which discusses resolution of Actions/Licensee Action Items (A/LAIs) from the NRC staff final SE of MRP-227, Revision 0 (Ref. 16).

3.1.2 Changes to Recommended Inspections for Primary, Expansion, and Existing Program Components

In the TR, some changes were made to the component identification/nomenclature, applicability, examination method and frequency, examination coverage, the linked expansion component(s) (for primary components) or the linked primary component(s) (for expansion components). This information is contained in Tables 4-1 through 4-9 of the TR.

Part 2 of Appendix C of the TR provides a line-by-line comparison of each line item from Table 4-1 through 4-9, comparing the MRP-227-A line item to the corresponding item in MRP-227, Revision 1. Since many of these changes were editorial, this section of the SE will discuss only those changes that the NRC staff considers significant.

3.1.3 Specific Line Item Changes

3.1.3.1 Babcock and Wilcox Primary Components – Table 4-1

For most or all primary components, the schedule for the initial (baseline) examination changed from "during the next 10-year ISI [in-service inspection program]" to "during the next 10-year ISI interval." In RAI 1, the NRC staff requested EPRI to clarify the meaning of "during the next 10-year ISI". The language is unclear as to whether this means the examination is to be performed during the next scheduled 10-year ISI examination of the reactor vessel internals, or sometime during the next 10-year ISI interval. If the latter is the case, the NRC staff was concerned that the language could allow these examinations to be performed as far in the future as 20 years from now, if the current 10-year ISI interval started today.

In its October 16, 2017, response to RAI 1 EPRI stated that the updated wording in MRP-227, Revision 1 does not allow initial examinations 20 years into the period of extended operation. EPRI further stated that per Section 4 on page 4-2 of MRP-227, Revision 1, the term "10-year ISI interval" is intended to mean the plant's existing schedule associated with removal of the core barrel, i.e., during the next scheduled 10-year ISI interval examination. EPRI stated that as defined in the ASME B&PV Code Section XI, the first inspection interval is defined as 10 years following the initial start of plant commercial startup, and successive inspection intervals are defined as 10 years following the previous inspection interval.

EPRI further stated that each inspection interval may be extended by as much as one year and may be reduced without restriction, provided the examinations required for the interval have been completed, and that successive intervals shall not extend more than 1 year beyond the original pattern of 10-year intervals and shall not exceed 11 years in length. EPRI stated that, therefore, for the baseline (i.e., initial) examinations, the intention of this wording is for examinations to be performed prior to the end of the fourth ASME ISI interval and not more than 11 years since the previous ASME ISI interval was completed, i.e., what is allowed by Section XI of the ASME B&PV Code. EPRI also stated that this is also consistent with the stipulations stated in Section 4.2.6 of MRP-227, Revision 1 for subsequent examination intervals.

In its May 17, 2018, letter, EPRI provided a new Note 10 would be added to Table 4-1 which states:

> The term “10-year ISI interval” is intended to mean the plant’s existing schedule associated with removal of the core barrel, i.e., during the next scheduled 10-year ISI interval examination. As defined in the ASME Boiler and Pressure Vessel (B&PV) Code Section XI, the first inspection interval is defined as 10 years following the initial start of plant commercial startup, and successive inspection intervals are defined as 10 years following the previous inspection interval. Each inspection interval may be extended by as much as 1 year and may be reduced without restriction, provided the examinations required for the interval have been completed. Successive intervals shall not extend more than 1 year beyond the original pattern of 10 year intervals and shall not exceed 11 years in length.

The staff understands from EPRI’s response to RAI 1, as clarified by the added note, that the intent of the wording is that the initial examinations be performed prior to the end of the fourth ISI interval. Based on the restrictions on extension of ISI intervals in Section, the initial baseline examinations would occur early in the PEO. RAI 1 is thus resolved.

In MRP-227, Revision 1, the examination coverage for the "Plenum Coverage Assembly & Core Support Shield Assembly” changed from “Determination of differential height of top of plenum rib pads to reactor vessel seating surface, with plenum in reactor vessel” to “Determination of differential height of top of plenum rib pads/plenum cover support ring location to reactor vessel seating surface, with plenum in reactor vessel.”

The change to this item was to add the plenum-cover support ring as a subcomponent and to add this subcomponent as an additional possible reference point for the physical measurement. The plenum-cover support ring appears to be a new subcomponent added in MRP-227, Revision 1. The plenum-cover support ring is addressed in MRP-189, Revision 1, “Materials Reliability Program: Screening, Categorization, and Ranking of B&W [Babcock and Wilcox]-Designed PWR Internals Component Items (Ref. 25)” and was determined to screen out for all degradation mechanisms, which was confirmed by the failure mode, effects and criticality analysis (FMECA).

The plenum-cover assembly – weldment rib pads and plenum-cover assembly – support flange were determined to screened in for wear and have moderate-to-high safety risk in MRP-189, Revision 1. Therefore, in RAI 2, the NRC staff requested the MRP clarify why the plenum-cover support ring was added as a subcomponent and how and why the support ring was added as a reference location for making the physical measurements.

The October 16, 2017, EPRI response to RAI 2 clarified that the plenum-cover support ring was inadvertently omitted from the description of the measurement reference point in MRP-227-A, and in fact, is machined on a common plane with the plenum rib pads. A sketch showing the

interfacing components, including the plenum-cover support ring and rib pads, is included with the response to RAI 2. The EPRI response also indicated that MRP-189, Revision 2 and MRP-231, Revision 3 "Materials Reliability Program: Aging Management Strategies for B&W Pressurized Water Reactor Internals" (Ref. 9), have been updated to reflect this correction.

The NRC staff reviewed the response to RAI 2 and finds that EPRI has clarified why the plenum-cover support ring was added as part of the reference point for the physical measurement. The NRC staff also reviewed MRP-189, Revision 2, "Materials Reliability Program: Screening, Categorization and Ranking of Babcock & Wilcox-Designed Pressurized Water Reactor Internals Component Items and Welds, (Ref. 8), MRP-231 Revision 3 and MRP-227, Revision 1, and finds that EPRI has appropriately updated the supporting FMECA and final characterization of the plenum-cover support ring (The plenum-cover support ring is now a primary component for wear as are the plenum-cover weldment rib pads). RAI 2 is thus resolved.

The control rod guide tube (CRGT) spacer castings previously had no expansion link. An expansion link to the vent valve bodies has now been added. The vent valve bodies were not an expansion component in MRP-227-A. According to MRP-189, Revision 1 (Ref. 25), the vent valve bodies are cast austenitic stainless steel (CASS), as are the CRGT spacer castings. Since the vent valve bodies were previously a no additional measures component, in RAI 3 the NRC staff asked why the vent valve bodies were made an expansion component for the CRGT spacer castings.

In its October 16, 2017, response to RAI 3, EPRI explained that thermal embrittlement (TE) was screened out for the vent valve bodies based on ferrite content. However, it was discovered that the vent valve bodies had been replaced with spare vent valves at some plants, and the certified material test reports (CMTRs) had not been located for the spare valves. Therefore, the vent valve bodies were considered potentially susceptible to TE and added in MRP-227, Revision 1 as an expansion component.

The vent valves were added as an expansion component for TE since, as stipulated in the response to RAI 3, the CRGT spacer castings are made from CASS material with high molybdenum (CF3M), which tends to have higher ferrite, while the vent valve bodies are low-molybdenum Type CF8 CASS which tends to have lower ferrite. High ferrite CASS materials are more susceptible to TE than low ferrite CASS materials. A note was added to Table 4-4 of MRP-227, Revision 1 that allows licensees the option of screening out the vent valve bodies using CMTR data, or using generic technical report PWROG-15032-NP, "Statistical Assessment of PWR RV Internals CASS Materials" (Ref. 26) to show that the vent valves have a high probability of having ferrite below the screening criterion for TE.

The NRC staff finds EPRI's explanation for the change to add the vent valve bodies as an expansion component linked to the CRGT spacer castings to be reasonable, since both are susceptible to TE, with the CRGT spacer castings likely to lead the vent valves in susceptibility due to higher molybdenum content. RAI 3 is thus resolved.

Baffle-to-former (B-F) Bolts – For the B&W B-F bolts, the schedule for the initial (baseline) ultrasonic testing (UT) examination changed from "no later than two refueling outages from the beginning of the license renewal period" to "volumetric (UT) examination during the next 10-year ISI interval." Since it is not clear when the next 10-year ISI interval starts (it could be up to ten years from the current date), the staff was concerned that this could result in the baseline examination being significantly later than two refueling outages from the beginning of the license renewal period.

It was not clear to the NRC staff whether this change assumes all six operating B&W units have already completed baseline UT examinations. Therefore, in RAI 4, the NRC staff asked EPRI whether the initial baseline UT examination schedule for the B-F bolts in MRP-227, Revision 1 assumed an examination of baffle-to-former bolts has been completed within two refueling outages from the beginning of the period of extended operation; or if not, to justify the change in the schedule for the initial (baseline) UT examination of the B-F bolts.

In its October 16, 2017, response to RAI 4, EPRI stated that the initial baseline UT examination schedule for the B-F bolts in MRP-227, Revision 1 (Reference 1) assumes an examination of B-F bolts has been completed within two refueling outages from the beginning of the PEO for each operating B&W unit. The NRC staff finds that the EPRI response is acceptable because it has verified that initial baseline examinations of B-F bolts will occur within two refueling outages of the start of the PEO.

In its May 17, 2018, clarification of RAI 4, EPRI indicated it would add Note 11 to the Examination Method/Frequency for B-F bolts that states:

> 11. This assumes that all units operating as of December 2011 have performed baseline (initial) volumetric (UT) examinations no later than two refueling outages from the beginning of their first license renewal period.

The NRC staff finds the addition of the note to be acceptable in addressing RAI 4. RAI 4 is thus resolved.

The lower grid shock pad bolts for Three Mile Island (TMI)-1 were moved from expansion to primary. Appendix C on p. C-17 indicates this change was the result of a plant-specific review. The NRC staff notes shock pad bolts for TMI-1 are made from a different material (Alloy X-750) than in the other B&W plants (Type 304 stainless steel), so in MRP-190 were determined to have a higher likelihood of stress-corrosion cracking (SCC). The NRC staff therefore finds this change to be acceptable due to the higher susceptibility of the TMI-1 bolt material to SCC.

In Table 4-1, or Item B15. “IMI [Incore Monitoring Instrumentation] Guide Tube Assembly Spiders and Spider welds,” – the examination coverage changed from “100 percent of top surfaces of 52 spider castings and welds to the adjacent lower grid rib section” in MRP-227-A to “Spiders: 100 percent of the accessible top surfaces and 100 percent of the accessible spider surfaces adjacent to the spider casting welds” and “Spider welds: 100 percent of the accessible welds to the adjacent lower grid rib section.” Therefore, in RAI 25 the NRC staff requested that EPRI explain why the description of the examination coverage for this item changed, and explain the significance of this change.

EPRI’s October 16, 2017, response to RAI 25 stated that the description of the examination coverage was updated to clarify the examination coverage by separating the IMI guide tube spiders and the welds to identify the specific areas of concern. The addition of the words “accessible” throughout the examination coverage description for these items have been added to maintain consistency with the examination coverage descriptions of other components items and welds in MRP-227, Revision 1. The NRC staff finds the EPRI response to RAI 25 acceptable because it clarifies the reason for the change in the wording. The NRC staff finds the revised description of the required examination coverage for the IMI guide tube spiders and welds is clearer than the previous wording. RAI 25 is thus resolved.

3.1.3.2 B&W expansion components

In Table 4-1, Item B11., “Core Barrel Assembly – Locking Devices,” including locking welds, of B-F bolts and internal baffle-to-baffle (B-B) bolts, has applicable aging mechanisms of irradiation assisted stress corrosion cracking (IASCC), IE including the detection of missing,

non-functional, or removed locking devices or welds, and has as an expansion link “locking devices, including locking welds, of the external B-B bolts and core barrel-to-former bolts.” However, in MRP-227, Revision 1, a new Note 8 has been added for the expansion link, which states that “the aging degradation mechanism of IASCC is only applicable to the baffle-to-former bolt and internal B-B bolt locking devices, not the B-F bolt and internal B-B bolt locking device welds.” However, under the expansion link column in Table 4-1, the expansion link for Item B11 is described as locking devices, including locking welds, of the external B-B bolts and core B-F bolts.

In RAI 6, the NRC staff therefore requested the following information:

a. Clarify whether the expansion link column or Note 8 is correct.
b. If Note 8 is correct, explain why IASCC is not applicable to the locking device welds, and why there are no expansion links for the welds.

In its October 16, 2017, response to RAI 6, Item a, EPRI stated that Note 8 in Table 4-1 and the expansion link in MRP-227, Revision 1 (Reference 1) are both correct. EPRI further stated that as discussed in Section 3.2.3 and stated in Note 1 of Table 3-3 in MRP-231, Revision 3 (Ref. 8), the locking devices for the B-F bolts and internal B-B bolts are primary component items for IASCC with no expansion link, and that therefore, Note 8 of Table 4-1 in MRP-227, Revision 1, is correct as written. EPRI also provided a proposed change to the “Effect (Mechanism)” column of Table 4-1 for Item B11., “Core Barrel Assembly—Locking devices, including locking welds, of baffle-to-former bolts and internal B-B bolts” in MRP-227, Revision 1 to show that IASCC is not applicable to the locking welds. EPRI stated that the reference notes would remain the same as those currently in MRP-227, Revision 1 Table 4-1.

The EPRI response to Item b of RAI 6 indicated that, based on the IASCC screening criteria of MRP-175, "Materials Reliability Program: PWR Internals Material Aging Degradation Mechanism Screening and Threshold Values” (Ref. 27), IASCC does not apply to the locking welds because they are not highly-stressed components. The only category of weld considered highly stressed by EPRI are multi-pass welds, which the locking welds are not.

The NRC staff understands from the response to RAI 6 that the expansion link from the B-F and internal B-B bolt locking devices to the locking devices of the external B-B bolt and core B-F bolts, and their associated welds, is for IE only.

The staff finds EPRI’s response to RAI 6 acceptable because it clarified why EPRI believes the locking device welds are not susceptible to IASCC, and why there is no expansion link for IASCC of the locking welds. More importantly, the staff also notes that the locking welds associated with the locking devices of the baffle-to-former bolts and internal B-B bolts will still be inspected for cracking, whether caused by IE or IASCC. RAI 6 is thus resolved.

The Upper Thermal Shield (UTS) bolts and surveillance specimen holder tube (SSHT) studs/nuts (CR-3) or bolts (DB) and their locking devices (both subcomponents of the Core Barrel Assembly), had changes to the “Effect (mechanism)” information. Specifically, irradiation-assisted stress relaxation (IC/ISR), wear, and fatigue were added for the SSHT bolts. In RAI 11, Item a, the staff asked EPRI to explain why the aging mechanism for the SSHT bolts was changed.

Note 7 to table 4-4 indicates that this table entry for the SSHT bolts also includes the aging degradation mechanisms of IC/ISR, wear and fatigue for the compression collar and washer for the SSHT bolt. The compression collars for the SSHT bolt are not included in the screening and FMECA documented in MRP-189, Revision 1 (Ref. 25) and MRP-190, "Materials Reliability Program: Failure Modes, Effects, and Criticality Analysis of B&W-Designed PWR Internals”

(Ref. 28). Therefore, in RAI 11, Item b, the NRC staff asked EPRI to clarify whether the compression collars were left out of the screening and FMECA process as an oversight, or whether the compression collars are the same as the SSHT bolt locking cups and tie plates that are included in the screening and FMECA. If the latter, the NRC staff asked in RAI 11 why the screening and FMECA results for these components changed.

In its October 16, 2017, response to RAI 11, EPRI indicated that the aging mechanisms of IC/ISR were added for the SSHT bolts because these bolts exceed the screening limit of $1.3x10^{20}$ neutrons per square centimeter (n/cm2) for IC/ISR, and wear and fatigue also are applicable for bolts that screen in for IC/ISR. The EPRI response to RAI 11, Item b indicated that Davis-Besse is the only plant with SSHT bolts, and that the original SSHT bolts at Davis-Besse have been replaced. EPRI further indicated that the records for the replacement bolts were found in 2010, after publication of MRP-189, Revision 1, which identified that the design of these SSHT bolts at Davis-Besse utilizes a bolt, compression collar, spherical washer, and a tie-plate/crimp locking cup assembly. RAI 11 is thus resolved.

In addition, for TMI-1 only, an additional primary link was added for the UTS bolts and SSHT bolts and their locking devices. The new link is to the shock pad bolts and their locking devices. The shock pad bolts for TMI-1 are made from a different material (Alloy X-750) than in the other B&W plants (Type 304 stainless steel), so in MRP-190 they were determined to have a higher likelihood of SCC. The staff finds it acceptable that the lower grid assembly – shock pad bolts have been moved from expansion to primary. The staff finds the selection of the expansion link appropriate since the UTS bolts and SSHT bolts are also Alloy X-750 bolts that may exhibit cracking due to SCC as an aging mechanisms. Additionally, the existing primary links for the UTS and SSHT bolts were maintained as well (i.e., UCB, LCB, and FD bolts), so no justification is needed for the shock pad bolts being a lead component.

For Table 4-4, B&W Plants expansion Items, Core Barrel Assembly, B11.1.Locking Devices, including locking welds, of the external B-B bolts and core barrel-to- former bolts, the primary link changed from:

"…locking devices, including locking welds, of baffle-to-former bolts **or** internal baffle-to- baffle bolts," to

"B11.Locking devices, including locking welds, of baffle-to-former bolts **and** internal B-B bolts.

Therefore, in RAI 17, the NRC staff asked EPRI if the change from "or" to "and" mean degradation now has to be exhibited in both the locking devices for the B-F bolts and the locking devices for the internal B-B bolts for the expansion to be required, whereas in MRP-227-A the expansion would be required if only one of these items exhibited degradation. If so, justify the changes.

In its October 16, 2017, response to RAI 17, EPRI stated that the change from "or" to "and" noted in this RAI does not mean degradation now has to be exhibited in both the locking devices for the baffle-to-former bolts and the locking devices for the internal B-B bolts for the expansion to be required. EPRI further stated that the primary link for the B11.1 expansion items was, and still is, required to be both types of locking devices, and that this change was editorial in nature only. In its May 17, 2018, letter, EPRI proposed revised wording to Table 4-4 Note 1 clarifying that degradation in either primary linked component triggers the expansion examination. The staff finds this clarification acceptable. RAI 17 is thus resolved.

The Lower Grid Assembly – Item B10.3. Lower Grid Rib Section has been added as an additional expansion link for primary Item B10. Core Barrel Assembly – Baffle Plates. Lower Grid Assembly – Item B10.3. Lower Grid Rib Section was not included in MRP-227-A as either a

primary or expansion component. In RAI 18, the staff asked why this item has apparently been recategorized from “no additional measures” to “expansion.”

EPRI’s October 16, 2017, response to RAI 18 explained that the lower grid rib section was recategorized to higher safety significance in the FMECA based on 1) screening in for IE based on a revised estimated neutron fluence; 2) its direct core support function; and 3) redundancy (multiple flaws would be needed to initiate and grow to critical flaw size in multiple ribs to cause a safety concern). Redundancy was a mitigating factor resulting in the lower grid rib section being assigned to a lower safety consequence category than if only the first two factors had been considered. The staff finds EPRI’s response to RAI 18 acceptable because it explains why the lower grid rib section was recategorized. Also, the staff finds the designation of the rib section as an “expansion” component rather than a primary component acceptable because it experiences less fluence than its primary link (i.e., baffle plates) and because of the redundant nature of its construction. RAI 18 is thus resolved.

3.1.3.3 B&W Plants Examination and Acceptance Criteria

In Table 5-1, "Primary Item Examinations Acceptance Criteria," the Core Barrel Assembly – Baffle-to-former bolts expansion criteria have changed. In MRP-227-A, the expansion criteria is “Confirmed unacceptable indications in greater than or equal to 5 percent (or 43) of the baffle-to-former bolts, provided that none of the unacceptable bolts are on former elevations 3, 4, and 5, or greater than 25 percent of the bolts on a single baffle plate, shall require an evaluation of the internal B-B bolts for the purpose of determining whether to examine or replace the internal baffle-to- baffle bolts. The evaluation may include external B-B bolts and core barrel-to-former bolts for the purpose of determining whether to replace them.” In MRP-227, Revision 1, the expansion criteria is “Confirmed unacceptable indications in greater than or equal to 5 percent of the baffle-to-former bolts (including previously failed/removed bolts) shall require an evaluation of the B-B bolts and the core barrel-to-former bolts by the completion of the next refueling outage. The evaluation shall also assess functionality of the core barrel assembly with aging degradation of the B-B bolts and core barrel-to-former bolts for the purpose of determining continued operation or replacement.”

The criteria requiring expansion if greater than 25 percent of the bolts on one baffle plate are degraded would result in expansion if clustering of degraded bolts was present, which has been seen in recent operating experience (OE) with baffle-former bolt (BFB) degradation in Westinghouse-design RVI. It is also not clear why the language regarding bolts on former elevations 3, 4, and 5 has been removed from the expansion criteria.

Therefore, in RAI 21, the NRC staff requested that EPRI provide the technical basis for the changes to the expansion criteria for the baffle-to-former bolts in B&W plants. The response should address the following items:

a) An explanation for the removal of the language from the expansion criteria related to bolts on former levels 3, 4, and 5, and whether this results in less conservativism. If less conservative, provide a justification for the reduction in conservatism.

b) Why was the expansion criterion of more than 25 percent of the bolts on a single plate [degraded] removed in Revision 1 especially considering recent OE with clustered BFB degradation?

In its October 16, 2017, response to RAI 21, Item a, EPRI stated that the expansion criteria were updated to only include consideration that an active age-related degradation mechanism in the B-F bolts would be present, as aging degradation drives the expansion inspections. EPRI

further stated that since the presence of the B-F bolts on former levels 3, 4, and 5 is a consideration for continued operation, and not expansion, associated language was removed from the expansion criteria related to the B-F bolts. EPRI also stated that as this language is not associated with potential core barrel-to-former (CB-F) bolt, internal B-B bolt, or external B-B bolt failures as predicted by B-F bolt failures, removal of the language does not result in less conservatism. Based on EPRI's response to RAI 21, item a, the staff understands that the B-F bolts on former levels 3, 4, and 5 are needed for operability, but do not represent an appropriate expansion criteria because degraded B-F bolts on any former level could indicate the potential for degradation in the expansion bolt components (i.e., CB-F bolts, B-B bolts). Therefore, the staff finds the removal of this expansion criteria in MRP-227, Revision 1, to be acceptable.

EPRI's October 16, 2017, response to RAI 21, Item b, indicated that unlike in Westinghouse-design RVI, clustering was not expected to occur in B&W plant B-F bolts because the B&W plants operate in an upflow configuration and bolting installation and design characteristics make the B-F bolts less susceptible to cracking. Current OE supports the assertion that B&W plant B-F bolts are less susceptible to cracking than Westinghouse B-F bolts.

The EPRI response provides more detail on the assessment of B&W B-F bolt susceptibility to clustered IASCC failure. EPRI further stated in their response that removal of the expansion criterion of more than 25 percent of the bolts on a single plate is appropriate because a Framatome evaluation has determined for the B&W design that clustered failures of B-F bolts on a given baffle plate have a negligible impact on the likelihood of failure of the associated CB-F bolts. The staff understands EPRI's explanation for removing the 25 percent clustering criterion and finds its removal acceptable because the expansion criteria are intended to determine when degradation may be anticipated in the expansion component while the previous 25 percent clustering criterion only indicated the extent of degradation in the B-F bolts and associated operability considerations. RAI 21 is thus resolved.

In Table 5-1, the examination acceptance criteria and expansion criteria for the Core Barrel Assembly – Baffle Plates have changed. In MRP-227-A, the examination acceptance criteria column in Table 5-1 stated that the specific relevant condition is readily detectable cracking in the baffle plates. In MRP-227, Revision 1, this has been changed to state the specific relevant condition is readily detectable cracking connecting openings in the baffle plates (i.e., each bolt hole and flow hole).

With respect to expansion criteria, in MRP-227-A, the expansion criteria states:

> Confirmed cracking in multiple (2 or more) locations in the baffle plates shall require expansion, with continued operation of former plates and the core barrel cylinder justified by evaluation or by replacement by the completion of the next refueling outage.

With respect to expansion criteria, MRP-227, Revision 1, states that gross cracking (if confirmed) within one inch of a bolt or flow-hole location in the baffle plates shall require:

a) An evaluation of the former plates and the core barrel cylinder for the purpose of determining continued operation or repair/replacement by the completion of the next refueling outage. Alternatively, repair/replacement activities may be initiated based on results of a best effort former plate and core barrel cylinder examination.

b) That the Visual (VT-3) examination be expanded by the completion of the next refueling outage to include 100 percent of the accessible portions of the lower grid rib section heat-affected zones adjacent to the IMI guide tube spider-to-lower grid rib section welds.

The relevant condition now requires cracking connecting openings in baffle plates, rather than just detectable cracking. Also, the expansion criteria in MRP-227, Revision 1, seem inconsistent with the relevant conditions since the relevant conditions require linkage of openings by cracking, while the expansion criteria only seem to require cracking within one inch of an opening.

Therefore, in RAI 22, the NRC staff requested the following information:

a) Provide a technical justification for the change in the definition of the relevant condition for the baffle plates, specifically, the new requirement that the cracking link openings in the baffle plates.

b) Provide a technical justification for the change in the expansion criteria for the baffle plates.

c) Clarify whether expansion is only required if cracking links two or more openings or whether expansion would be required if cracking is present within one inch of any opening.

EPRI's October 16, 2017, response to RAI 22, Part a stated that the relevant condition from Table 5-1 of MRP-227-A will be retained in MRP-227, Revision 1. EPRI provided a markup showing the change. Therefore, EPRI removed the stipulation in the relevant condition that the cracking connect adjacent holes in Table 5-1 of MRP-227, Revision 1.

In its October 16, 2017, response to RAI 22, Part b, EPRI stated that the expansion criteria from Table 5-1 of MRP-227, Revision 1 were updated for two reasons: 1) previously, MRP-227-A required two instances of confirmed cracking before expansion; MRP-227, Revision 1, only requires one instance of confirmed gross cracking (within one inch of a bolt or flow hole, which is the examination coverage for both MRP-227-A and MRP-227, Revision 1), and 2) part a of the expansion criteria were updated to reflect Licensee/Applicant Action Item 6 from MRP-227-A and part b of the expansion criteria added the lower-grid rib section as an expansion item to the baffle plates as documented in the response to RAI 18 in Section 3.1.3.2 of this document. In response to RAI 22, Part c, EPRI stated that expansion is required if there is confirmed gross cracking within one inch of a bolt or flow hole location in the baffle plate.

The NRC staff finds the EPRI response to RAI 22, Item a acceptable because it proposes to remove the language in Table 5-1, for the baffle plates that required cracking to connect openings in the baffle plate. The EPRI response to RAI 22, Item b indicates the changes to part a of the expansion criteria were made to reflect A/LAI 6 from MRP-227-A. A/LAI 6 essentially requires that functionality of certain B&W RVI components that are inaccessible for examination, be justified by evaluation. The first piece of part a of the expansion criteria is consistent with A/LAI 6 in that it specifies an evaluation of the core-barrel cylinder and former plates or replacement, because these components are inaccessible for examination. The EPRI justification for Part b of the expansion criteria is acceptable because, as documented in the response to RAI-18, the lower-grid rib section has been added as a new expansion component and linked to the baffle plates which have higher fluence and are expected to exhibit IASCC sooner.

The staff notes that, with the changes proposed in the RAI 22 response, the expansion criteria for the baffle plates in MRP-227, Revision 1, are actually more conservative than the expansion criteria for MRP-227-A because the revised expansion criteria only require cracking at one location versus two. RAI 22 is thus resolved.

3.1.3.4 Combustion Engineering Primary Components

For Item C12: "Lower Support Structure – Deep Beams," the examination coverage has been changed to 25 percent of the total number of beam-to-beam welds. The examination coverage in MRP-227-A for the deep beams does not specify a percentage of beam-to-beam welds that must be examined, but it is implied that 100 percent of the welds should be examined. A similar change in coverage was made for Item C5.4., "Lower Support Structure – Lower Core Support Beams."

In RAI 10 the NRC staff a) requested EPRI to provide a justification for this apparent reduction in coverage, and b) asked what expansion to the remaining beam-to-beam welds would be conducted if degradation is found in the initial 25 percent sample. In its January 30, 2018, response to RAI 10, EPRI provided a justification for the reduction in coverage. The justification is based on the high level of redundancy in the deep-beam structure. There are multiple welds for each deep beam. Due to the high level of redundancy, multiple weld failures would be required to compromise function.

EPRI also indicated that the onset of loss of structural functionality would likely first be detected during fuel loading or unloading conducted during each refueling outage. EPRI indicated that stresses were expected to be higher near the outer edges of the assembly, but dose is highest near the center of the assembly. Therefore, EPRI proposed to modify the coverage requirement to require that the inspection be spread out across the structure and to be performed on different sets of welds after each inspection interval. EPRI provided a revision of the text for the "Examination Coverage" column for the deep beams in Table 4-2 of MRP-227, Revision 1.

With respect to the Lower Core Support Structure – Lower Core Support Beams, EPRI indicated these beams function to support the core-support columns (CSCs) and have similar redundancy as the CSCs. EPRI indicated that the evaluation in PWROG-14048-P, Revision 1, "Functionality Analysis: Lower Support Columns," (Ref. 43) showed that greater than 50 percent of the CSCs could be degraded without loss of function, thus it is expected that a similar level of degradation could be tolerated in the beams. EPRI stated that this margin to loss of functions provides the technical justification for the reduced coverage. EPRI also stated that the location and geometry of the lower-core support beams presents significant challenges to obtaining higher coverage levels and that coverage levels of 75 percent or 100 percent are likely not attainable.

In response to Item b, EPRI stated that if degradation is found in the initial 25 percent inspection population, the examination would be expanded to include the remaining beam-to-beam welds, and that the deep beam line item in Tale 4-2 will be updated to include a reference to the Existing Note 6 in the table, which specifies that the stated coverage requirement is the minimum if no significant indications are found. EPRI provided a markup of Table 5-2 that adds expansion criteria that require inspection of the remaining deep beams by the completion of the next refueling outage. In its letter dated May 17, 2018, EPRI provided additional supporting information justifying not requiring the expansion to be completed during the same refueling outage that degradation is initially found. The key points of the additional information are that the lower-support structure is highly redundant, thus function would be maintained even if an entire weld fails, and insertion and removal of fuel during each outage provides an element of regular monitoring which is expected to detect a loss of functionality.

Based on the above, EPRI's justification for the reduction in coverage for the deep beams and the lower core support beams is based on the structural redundancy of the lower support structure as demonstrated by PWROG-14048-P, Revision 1. In its publicly available staff assessment of PWROG-14048-P, Revision 1, the NRC staff generally found the conclusions of that report to be acceptable. EPRI modified the coverage description for the deep beams to

require the examinations be evenly distributed across the structure. In addition, insertion of fuel each outage provides regular monitoring of the functionality of the deep beam structure. The staff therefore finds EPRI has provided an adequate justification for the reduction in coverage, and RAI 10 is thus resolved.

In Table 5-2, "CE Plants Examination Acceptance and expansion Criteria," for the Core Shroud Assembly (welded) – Assembly, the examination acceptance criteria in MRP-227, Revision 1, specifies a VT- 3 examination but a VT-1 examination is specified in Table 4-2 for this item. MRP-227-A specified VT- 1 in both tables for this item. Therefore, in RAI 23, the NRC staff asked EPRI to clarify whether VT-1 or VT-3 is the intended technique, and if VT-3 is the intended technique, explain why this technique is acceptable to address the amount of physical separation expected if distortion is occurring.

The October 16, 2017, EPRI response to RAI 23 stated that the examination acceptance criteria in Table 5-2 for the Core Shroud Assembly (Welded) Assembly should specify a VT-1 examination, consistent with that specified in Table 4-2. This is also the same as what was required in MRP-227-A. Table 5-2 of MRP-227, Revision 1, will be updated. The RAI response provided a markup of Table 5-2 showing this change. Since EPRI corrected this inconsistency in the tables, the NRC staff finds that RAI 23 is resolved.

In MRP-227, Revision 1, Table 4-2, three Combustion Engineering (CE) primary components state under "Examination Method/Frequency," "If screening for fatigue cannot be satisfied by plant-specific evaluation, 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." The language for the corresponding components in MRP-227-A for "Examination scope/frequency" stated "If fatigue life cannot be demonstrated by time-limited aging analysis (TLAA), 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."

The components subject to the fatigue screening are C7., "Core Support Barrel Assembly – CSB [Core Support Barrel] Flexure Weld (CSBFW)," C9., "Lower Support Structure – Core Support Plate," and C10., "Upper Internals Assembly – Fuel Alignment Plate." Also, in Table 4.2, for Item C7., SCC has been added as a degradation mechanism yet the examination method allows examination to be avoided provided the item passes a screening for fatigue.

Therefore, in RAI 16 the NRC staff requested EPRI to:

a) Define and justify the criteria that are to be used for screening for fatigue. Is a specific cumulative usage factor (CUF) value used as a screening criterion? Are environmental effects to be considered? If so, how are environmental effects to be included in the evaluation? EPRI should also discuss whether such a criterion should be added to Table 4-2.

b) Justify how fatigue screening accounts for possible SCC contributions for Item C.7. Is additional evaluation or inspection of the CSBFW needed to address possible SCC?

In its January 30, 2018, response to RAI 16, Item a, EPRI stated that the fatigue screening criterion that is provided in MRP-175, Revision 0, which was used in the development of MRP-227-A, is applied for screening for fatigue. EPRI also stated that MRP-175, Revision 0 utilizes a screening CUF of 0.1 at 40 years, which was intended to address potential environmental effects, and that since environmental effects were considered in the MRP-175, Revision 0 screening, there is not a need to add a separate criterion related to it in Table 4-2 of MRP-227, Revision 1.

In response to RAI 16, Item b, EPRI stated that regarding the degradation mechanisms of the CSBFW, fatigue screening does not account for possible contributions from SCC. EPRI stated that provided that the component does not screen-in for fatigue, an inspection would need to be performed to confirm there is no material degradation resulting from SCC, or an evaluation could be performed, using plant-specific or bounding information, in place of inspecting the CSBFW for effects of SCC. EPRI provided a revision to Table 4-2 of MRP-227, Revision 1, showing the change.

The NRC staff finds the EPRI response to RAI 16, Item a acceptable because it clarifies the criteria used to screen the CE primary components for fatigue. The NRC staff finds it acceptable that environmental effects were not explicitly considered because the screening criteria of a CUF of 0.1 is one-tenth of the ASME Code acceptance criterion for acceptable fatigue usage of ≤ 1.0. Therefore, this would accommodate an environmental effect of up to a multiplier of ten on the fatigue usage without causing the ASME Code criteria for fatigue to be exceeded. The NRC staff finds the EPRI response to RAI 16, Item b acceptable because it modified the examination method/frequency to address SCC. RAI 16 is thus resolved.

The NRC staff understands that the acceptance criterion for plant-specific fatigue evaluations may differ from the generic fatigue screening criterion of a non-environmentally adjusted CUF less than or equal to 0.1, because the plant-specific fatigue acceptance criteria are based on the plant-specific design and licensing basis.

3.1.3.5 *CE Expansion Components*

CE Core Shroud Welds (expansion)

In Table 4-5, for plant designs with core shrouds assembled with full-height shroud plates, the core shroud assembly, remaining axial welds, ribs and rings has been split into two items: C3.1, "Remaining axial welds," and C3.2, "Ribs and rings." The coverage for these two items is different, 75 percent for the remaining axial-weld length and adjacent base metal as visible from the core side of the shroud other than that already inspected under the primary link, and 25 percent of the ribs and rings.

Also, in MRP-227, Revision 1, Core Shroud Assembly (Welded) Item C2.1, "Remaining Axial Welds," is a new expansion component applicable to plant designs with core shrouds assembled in two vertical sections. The coverage for Item C2.1 is the same as for Item C3.1. In MRP-227-A, the coverage for the axial welds, ribs and rings was "axial welds seams" other than the core shroud reentrant corner welds at the core mid- plane, plus ribs and rings. Although the extent of coverage required has been quantified, no justification is provided for the examination coverages for the remaining axial welds, or the ribs and rings.

Also, in Figure 4-37, it is not clear if the core shroud assembly can be removed from the core support barrel assembly to allow examination of the ribs and rings. Therefore, in RAI 12, the NRC staff requested the following:

a) For Item C2.1 and 3.1, does 75 percent of the remaining axial weld length for the remaining axial welds mean a minimum of 75 percent of the total accessible plus inaccessible length of these welds must be examined to claim examination credit?
b) Justify the 25 percent sample size for the ribs and rings (Item C3.2).
c) Clarify whether the ribs and rings are accessible for visual examination.

In its January 30, 2018, response to RAI 12, Item a, EPRI clarified that the intended coverage for the remaining axial welds is 75 percent of un-inspected weld length that is visible on the core side of the shroud, including both inaccessible and accessible portions of the weld length. However, EPRI stated it expects most or all of the weld length to be accessible. In response to Items b and c, EPRI indicated that it has determined the ribs and rings are inaccessible, and that the "Examination Method/Frequency" in Table 4-5 for the ribs and rings will be modified to "justify by evaluation or replacement." EPRI provided a markup of Table 4-5 showing the change.

The NRC staff finds the EPRI response to RAI 12 acceptable because it clarifies the coverage requirements for the CE expansion core shroud welds, ribs and rings. The NRC staff finds the approach to justify by evaluation or replacement to be acceptable for the ribs and rings since these items are inaccessible. RAI 12 is thus resolved.

In MRP-227, Revision 1, Table 5-2, the expansion criterion for the Upper Flange Weld (UFW) requires inspection of the upper girth weld (UGW), lower girth weld (LGW), and Upper Axial Weld (UAW), by the completion of the next refueling outage, and to the lower core support beams within the next three refueling outages. In MRP-227-A, for the corresponding item in Table 5-2, the Core Support Barrel Assembly – Upper (core support barrel) flange weld, the expansion to the lower core support beams was required by the completion of the next refueling outage. In RAI 24, the NRC staff asked EPRI for the technical basis for changing the time frame for the expansion inspection of the lower-core support beams to within the next three refueling cycles.

The EPRI October 16, 2017, response to RAI 24 indicated the change was based the high degree of structural redundancy in the lower-core support beams, and also that one refueling cycle may not be enough time to develop the tooling and procedures to perform the examination.

The NRC staff reviewed MRP-191, Revision 1, "Materials Reliability Program: Screening, Categorization, and Ranking of Reactor Internals Components for Westinghouse and Combustion Engineering PWR Design" (Ref. 13) and verified the consequence of failure of the core support beams is low. The staff also finds that the core-support beams are redundant. Therefore, the staff finds EPRI has acceptably justified the change in the timing of the expansion examination of the core-support beams. RAI 24 is thus resolved.

3.1.3.6 CE Plants Examination and Acceptance Criteria

The changes to the CE plants examination and acceptance criteria in Table 5-2 are reflective of the changes to Tables 4-2 and Table 4-5 and are therefore acceptable.

3.1.3.7 Westinghouse Primary Components

Internals Hold Down Spring

For Alignment and Interfacing Components – Hold Down Spring, under Examination Method/Frequency, MRP-227-A states that if the first set of measurements is not sufficient to determine life, spring height measurements must be taken during the next two outages, in order to extrapolate the expected spring height to 60 years. MRP-227, Revision 1, states that if the first set of measurements is not sufficient to assess remaining life, additional spring height measurements will be required. Although the revised requirement is less prescriptive, the staff finds the revised examination frequency to be acceptable because it may be possible to show acceptable spring height beyond the two outages subsequent to the initial measurement, even if it is not possible to show acceptable spring height until the end of life.

Baffle-Former Bolts

OE in 2016 showed that Westinghouse 4-loop design plants operating in a downflow configuration with Type 347 stainless steel BFBs experienced higher-than-expected levels of degradation of BFB, and also significant clustering of degraded bolts. However, MRP-227, Revision 1, did not include any changes in the guidance for BFB from MRP-227-A.

Westinghouse Nuclear Safety Advisory Letter (NSAL)-16-1, Revision 1, "Baffle-Former Bolts" (Ref. 29) categorized all Westinghouse and CE design RVI with respect to susceptibility to BFB degradation. EPRI interim guidance in MRP Letter 2016-021 (Ref. 30), transmitted to NRC in MRP Letter 2016-022 (Ref. 31) endorsed the recommendation of Westinghouse NSAL16-1 that 4-loop, downflow plants with Type 347 bolts complete baseline UT examinations of BFB by the next refueling outage. These baseline examinations were completed by the end of 2017.

EPRI interim guidance in MRP Letter 2017-009, dated March 15, 2017 (Ref. 32), transmitted to the NRC via letter MRP 2017-011 (Ref. 33), accelerated the initial UT examination timing for 2-loop and 3-loop downflow plants (Tier 2) and specified limits on the maximum timing for subsequent examination for all plants with BFBs as a function of flow configuration (upflow or downflow) and the percentage of the total BFB population with indications, in addition to incorporating the guidance for Tier 1 plants from MRP Letter 2016-021. The EPRI interim guidance does not provide any guidance on how the subsequent examination interval is to be determined for Tier 1 and Tier 2 plants. The default subsequent examination interval in MRP-227, Revision 1, remains 10 years. However, the NRC staff was concerned that a default subsequent examination interval of ten years may not be appropriate for the highest susceptibility groups of plants.

If BFB degradation is found, an engineering evaluation is required. MRP-227, Revision 1, Section 7.5 defines Nuclear Energy Institute (NEI) 03-08, "Guideline for The Management of Materials Issues," "needed" guidance that, examination results that do not meet the examination acceptance criteria defined in Section 5 of these guidelines shall be recorded and entered in the plant corrective-action program and dispositioned, and that such engineering evaluations shall be conducted in accordance with NRC-accepted evaluation methods (i.e., ASME Code Section XI, WCAP-17096-NP or equivalent method). Current NRC-accepted guidance for determining the subsequent examination interval for BFBs is found in WCAP-17096-NP-A, Revision 2 "Reactor Internals Acceptance Criteria Methodology and Data Requirements" (Ref. 24), pages E-42 to E-43, which allows a subsequent examination interval of 10 years provided that no more of 50 percent of the initial margin with respect to the minimum required number of bolts is found degraded at the initial UT examination.

However, WCAP-17096-NP-A does not provide guidance for determining the subsequent examination interval if greater than 50 percent of the bolts constituting the margin are degraded, even if degraded bolts are replaced. In addition, the guidance in WCAP-17096-NP-A for determining the subsequent examination interval does not take into account the possibility of clustering of degraded bolts as was seen in the four-loop plants in 2016, and did not account for the large extent of BFB degradation seen in certain plants.

Therefore, in RAI 8, the NRC staff requested that EPRI:

a) Discuss whether revised guidance for BFB needs to be incorporated into MRP-227, Revision 1. If not, why not?

b) If such guidance should be incorporated, provide specifics on the initial examination coverage and schedule, and on how the subsequent examination coverage and timing would be determined.

c) Considering the recent OE with BFB degradation, justify that a 10-year subsequent examination interval remains appropriate for BFB. This justification should consider the possible effects of clustering.

d) How will the schedule for subsequent examination be determined if examination results show that greater than 50 percent of the numerical margin of bolts is degraded?

e) Provide a justification that the criteria allowing subsequent examination of BFB may be performed in 10 years, provided 50 percent or less of the numerical margin of BFB is degraded, is still appropriate considering the discovery of clustering of degraded BFB, and the discovery of more extensive BFB degradation than expected.

EPRI's response to RAI 8, Items a and b indicated that the guidance of MRP Letter 2016-021 and MRP Letter 2017-009 with regard to initial examination coverage and schedule and subsequent examination coverage and timing would be incorporated into the final version of MRP-227, Revision 1.

In response to Item c of RAI 8, EPRI indicated that limitations have been placed on the ten-year re-examination period if atypical or aggressive BFB degradation has been observed. Ten years is an upper limit for re-inspection interval and the exact re-inspection interval must be justified by a plant-specific evaluation if it exceeds the following defined limits. EPRI stated that for downflow plants with indications in three percent or greater of the bolt population, for upflow plants with indications in five percent or greater of the bolt population, or for downflow or upflow plants that demonstrate clustering of indications, the re-inspection period is not to exceed six years.

EPRI further stated that clustering is defined in MRP 2017-009 as three or more adjacent defective BFBs or more than 40 percent defective BFBs on the same baffle plate. EPRI stated that this re-examination period can be extended to 10 years through a plant-specific evaluation that justifies such an extension, and that these defined limits on re-inspection periods, in addition to being referenced in an updated MRP-227, Revision 1, will be incorporated into a revised version of WCAP-17096-NP-A, Revision 2 (possibly through Interim Guidance). EPRI's revision of Note 12 to Table 4-3 in its May 17, 2018, letter captures the definition of atypical or aggressive degradation and the requirements for plant-specific evaluation of such degradation.

The EPRI response to RAI 8 Items d and e can be summarized as follows: WCAP-17096-NP-A, Revision 2 is still the applicable document for addressing BFB margin requirements, and MRP-227, Revision 1 will still reference that methodology report. However, EPRI also indicated that a PWR Owners Group (PWROG) program is underway to make changes to the acceptance criteria methodology document consistent with the BFB interim guidance issued via EPRI letters MRP 2016-021 and MRP 2017-009. Specifically, the updates will include the re-inspection criteria (based on percent of UT indications and the presence of clustering, as described in the response to part C) and associated re-examination intervals.

The EPRI response to RAI 8 also included a markup of Table 4-3 of MRP-227, Revision 1, showing the changes to the "Examination Method/Frequency" column of the table for item W6. "Baffle-Former Assembly Baffle-Former Bolts." The baseline examination schedule was changed from 25 to 35 effective, full-power years (EFPY) to state "interval is dependent on the plant design (Note 11). Subsequent examination is dependent on the plant design and the results of the baseline inspection (Note 12)." The baseline examination schedule in Note 11 is the same as that specified in MRP 2017-009.

The NRC staff previously completed an assessment of EPRI Interim Guidance Letters MRP 2016-021 and MRP 2017-009 (Ref. 34). The assessment concluded that the guidance

with respect to initial examination schedules, and the maximum limits on subsequent examination intervals in EPRI MRP 2016-021 and MRP 2017-009, as modified by the responses to the staff's questions in EPRI's July 13, 2017, letter, provides acceptable aging management of BFBs in Westinghouse and CE designs.

The NRC staff conclusions that the changes to the initial examination schedules were acceptable were based primarily on its risk-informed evaluation of BFB degradation (Ref. 35), which concluded that UT examination at the next refueling outage resulted in acceptable risk for the most susceptible group of plants (Tier 1a), and its assessment of operating experience. Other tiers are less likely to have significant BFB degradation, based on OE. Moving up the initial examination schedule to 30 EFPY maximum is appropriate for Tier 2 plants, since these plants have found moderate levels of BFB degradation, but are at lower risk of significant degradation than Tier 1 plants.

With respect to the limits on subsequent examinations, the staff's conclusion in the staff assessment was based on review of Summary 'White Paper' of the Baffle-Former Bolt Prediction Results Provided by Structural Integrity Associates, AREVA, and Westinghouse, MRP Letter 2017-010 March 17, 2017 (Ref. 36), that describes three probabilistic models that were used to develop the limits on subsequent examination interval.

The staff finds that the EPRI response to Items a and b of RAI 8 is acceptable because the response explained that the interim guidance for BFB examinations will be incorporated into MRP-227, Revision 1. The response also described the changes to the initial examination schedule and explained how the subsequent examination interval is to be determined. The NRC staff found the EPRI response to Item c acceptable because it explains that limitations are placed on the subsequent examination interval, specifically a maximum of six years, if significant BFB degradation is detected in the initial examination. The guidance to be incorporated into MRP-227, Revision 1, is consistent with the interim guidance that has already been accepted by the staff in its staff assessment (Ref. 34).

The NRC staff finds EPRI's response to RAI 8 parts d and e acceptable because it clarifies that licensees must follow the guidance from MRP 2017-009 which will be incorporated in MRP-227, Revision 1. This limits the interval for subsequent examination based on the percentage of BFBs with indications in the initial examination and the plant configuration (downflow or upflow). RAI 8 is thus resolved.

EPRI's response to RAI 8 proposed that plant-specific evaluations to extend the subsequent examination interval beyond 6 years will be submitted to NRC for information within one year of discovery of the degradation that triggers a reduced reinspection interval, or if the evaluation is completed after this one-year timeframe it shall be submitted within 90 days of completion of the evaluation.

Since EPRI's response states that if the evaluation is completed after this one-year timeframe it shall be submitted within 90 days of completion of the evaluation, the staff was concerned that a licensee could complete its evaluation very close to the end of the shortened interval (e.g., 6 years), thus the 90 day time frame would not allow the staff sufficient time to review the evaluation.

The NRC staff assessment of EPRI's BFB interim guidance recommended the following:

> If the table in MRP 2017-009 indicates that the subsequent inspection interval is not to exceed 6 years (e.g., downflow plants with ≥ 3 percent BFBs with indications or clustering, or upflow plants with ≥ 5 percent of BFBs with indications or clustering), the plant-specific evaluation to determine a subsequent inspection interval should be submitted to the NRC for information within one year following the outage in which the

> degradation was found. Any evaluation to lengthen the determined inspection interval or to exceed the maximum inspection interval recommended in MRP-2017-009 should be submitted to the NRC for information at least one year prior to the end of the current applicable interval for BFB subsequent examination. This recommendation should be incorporated into the final NRC-approved version of MRP-227, Revision 1.

Therefore, in RAI 30, in order to ensure the NRC has sufficient time to review the BFB plant-specific evaluations, the staff requested EPRI to confirm that the above recommendation would be incorporated in MRP-227, Revision 1, or another NRC-accepted industry guidance document such as WCAP-17096-NP-A, or provide a basis for not doing so. In its September 28, 2018, response to RAI 30 (Ref. 6), EPRI stated it has issued interim guidance to its members in PWROG Report PWROG-17071, Revision 0 (Ref. 77), and that this topic is addressed in Section 2.3.2 of that report. PWROG-17071-NP, Revision 0 was submitted to the NRC for information via letter dated July 12, 2018 (Ref. 78). The guidance for evaluation of BFB degradation in PWROG-17071-NP, Revision 0 contains the following note:

> Note: Any plant-specific evaluation used to extend the re-inspection interval beyond those defined in MRP 2017-009 is to be submitted to the NRC for information at least one year prior to the end of the current applicable interval for BFB subsequent examination.

The staff observes that the guidance in PWROG-17071-NP, Revision 0, is not fully consistent with the NRC recommendation. However, EPRI also stated that it would incorporate the staff's recommendation related to submittal of BFB plant-specific evaluations to NRC into a revision of WCAP-17096-NP-A, which will be prepared in 2019. This will make the industry guidance consistent with the recommendation from the staff assessment of the BFB interim guidance. Since the revision to WCAP-17096-NP-A will not be submitted in time for the NRC to review it prior to the issuance of this SE, the NRC identifies as an applicant/licensee action item (A/LAI) that applicants and licensees shall submit BFB plant-specific evaluations in accordance with the NRC staff recommendation in the staff assessment of the BFB interim guidance. This is A/LAI 1 (Section 4.0). With the identification of A/LAI 1, RAI 30 is thus resolved.

<u>Control Rod Guide Tube Assembly Guide Plates (Cards)</u>

In MRP-227, Revision 1, the examination method/frequency was changed from visual VT-3 examination no later than 2 refueling outages from the beginning of the LR period, and no earlier than 2 refueling outages prior to the start of the LR period, to "per the requirements of WCAP-17451-P, "Reactor Internals Guide Tube Wear - Westinghouse Domestic Fleet Operational Projections" (Ref. 37) including subsequent examinations (Note 7)."

In addition, the examination coverage for guide cards has been changed from 20 percent examination of the number of CRGT assemblies, with all guide cards within each selected CRGT assembly examined, to "Examination coverage per the requirements of WCAP-17451-P, Revision 1 (Note 7)." Note 7 states:

> In WCAP-17541-P, Revision 1 the baseline examination schedule has been adjusted for various CRGT designs, the extent of individual CRGT examination modified, and flexible subsequent examination regimens correlating to initial baseline sample size, accuracy of wear estimation and examination results. Initial inspection prior to the license renewal period may be required.

The NRC staff reviewed the revised evaluation methodology for guide cards based on WCAP-17451-P, Revision 1, as part of its review of WCAP-17096-NP, Revision 2, for which the NRC staff SE was issued on May 3, 2016 (Ref. 38). WCAP-17096-NP-A, Revision 2, incorporating the NRC staff SE was published on August 31, 2016 (Ref. 23).

In the SE related to WCAP-17096-NP, Revision 2, the NRC staff found that the evaluation methodology and acceptance criteria for the guide cards based on WCAP-17451-P acceptable. The basis for this finding was that WCAP-17096-NP, Revision 2, provides a methodology for measuring wear that is based on ensuring functionality of the rod cluster control assemblies (RCCAs), and the acceptance criteria provide margin for future wear. In addition, the WCAP-17451-P report provides a rigorous and comprehensive basis for the methods and criteria for guide card wear evaluation.

With respect to examination coverage, Reference 56 notes that the coverage will be 76 to 87 percent depending on the reactor design, and that the inspection scope shall include at least the lower six guide cards per guide tube and, when needed, the top of the continuous guidance sheaths or C-tubes. In comparison, MRP-227-A, Table 4-3, specifies a scope of 20 percent examination of the total population of CRGTs, with all guide cards within each selected CRGT examined. The NRC staff notes that coverage based on WCAP-17451-P could be less on a per-CRGT basis than 100 percent (all guide cards) specified in MRP-227-A, but that a greater percentage of CRGTs will be sampled. Reference 34 indicates that the NRC staff reviewed OE information in WCAP-17451-P which it found provided adequate justification that only the bottom six guide cards need to be inspected since these tend to have the greatest amount of wear. Based on the above, the NRC staff finds the examination coverage for the guide cards acceptable.

With respect to the schedule for initial examinations, the revised methodology requires no wear measurements prior to 2015 and that alternate wear measurement schedules may be developed based on the guidance provided in WCAP-17451-P, Revision 1. This initial examination schedule is different than that specified in MRP-227-A, which specifies the initial inspection not later than 2 refueling outages after the start of the period of extended operation. The NRC staff notes that the change in guide card inspection requirements was communicated to the PWR licensees via letter MRP 2014-006, dated February 18, 2014 (Ref. 39).

Enclosure 1 to MRP Letter 2014-006 states that:

> …determination of when initial guide tube inspection measurements should be performed is based on a review of numerous foreign material examination videos of guide tube interiors performed at many plants as part of previous guide tube support pin replacement projects and from previous guide tube wear inspections performed for the PWROG. Results of the maximum wear per plant are provided. With these examination results the operational extension curves are used to predict when the first inspections should be performed.

On November 18, 2016, Westinghouse submitted a notification pursuant to 10 CFR Part 21, "Reporting of Defects and Noncompliance," that notified the NRC of a potential significant safety hazard due to guide card wear in four plants that use ion nitrided RCCAs in conjunction with 17x17 A or 17x17 AS style guide tubes (Ref. 40). Guide card wear in these plants may occur more rapidly than predicted by WCAP-17451-P, Revision 1, which is referenced in MRP-227, Revision 1 with respect to the examination schedule, method, and coverage for CRGT guide plates (guide cards) in Westinghouse-design RVI. Westinghouse NSAL 17-1 contains additional details on the operating experience with accelerated wear and recommended accelerated baseline examination schedules for the plants within scope of the 10 CFR Part 21.

Therefore, in RAI 19, the NRC staff requested that EPRI discuss how MRP-227, Revision 1 and/or WCAP-17451-P, Revision 1 should be modified to address the OE discussed in the 10 CFR Part 21 notification related to guide cards.

EPRI transmitted its interim guidance regarding guide cards to the NRC for information only via letter dated March 23, 2018 (Ref. 41). The interim guidance accelerates the baseline examination schedule for the plants with 17x17 A, 17x17 AS, and 17x17 AXLR[1] guide tubes (e.g., those addressed in the 10 CFR Part 21 notification) to either 2018 or 2020, except for Catawba, Unit 2, which already performed baseline guide card wear measurements in 2016, and plants with foreign material exclusion videos analyzed in Table 5-14 of WCAP-17451-P, which can perform the baseline guide card wear measurements according to the Table 5-14 schedule. The interim guidance also modified the recommended sample size for the baseline inspection scope. Sample sizes are provided as a function of number of loops and number of guide tubes with RCCAs for 95 percent or 99 percent confidence of detecting the guide tube with the second highest wear. This table supersedes Table 5-16 of WCAP-17451-P. The NRC staff notes that the minimum sample sizes are similar for the 95 percent level, but the WCAP-17541-P table provided sample sizes for lower confidence levels. The interim guidance recommends inspecting the sample size for 95 percent confidence as a minimum, which is more conservative than WCAP-17451-P.

In its May 17, 2018, letter, EPRI provided a clarification of the RAI 19 response, which states that the current applicable version of WCAP-17451-P is Revision 1, and that this revision is still applicable for many plants. EPRI further stated that recent OE has led to the creation of interim guidance on WCAP-17451-P, Revision 1 for certain control rod designs, and that this interim guidance has been issued as NEI 03-08 “Needed” guidance under PWROG letter OG-18-46 and was provided to the staff for information under PWROG letter OG-18-76.

EPRI also stated that for MRP-227, Revision 1, Note 7 will be revised to reference WCAP-17451-P, Revision 1 as modified by the interim guidance of letter OG-18-46, and that the revised note will state:

> In WCAP-17451-P Revision 1 the baseline examination schedule has been adjusted for various CRGT designs, the extent of individual CRGT examination modified, and flexible subsequent examination regimens correlating to initial baseline sample size, accuracy of wear estimation and examination results. Initial inspection prior to the license renewal period may be required. Use WCAP-17451-P, Revision 1, including the modified requirements due to the interim guidance provided in letter OG-18-46.

The NRC staff finds the revised response to RAI 19 acceptable because it refers to the interim guidance letter, and the interim guidance stipulates accelerated guide card examination scheduled for those plants with guide card designs addressed by the 10 CFR Part 21 notification. Therefore, the NRC staff finds that the proposed change to MRP-227, Revision 1, adequately addresses the OE related to accelerated wear of certain guide card designs. RAI 19 is thus resolved.

<u>Changes to Expansion Links</u>

For the core barrel assembly –LGW, the upper core plate and lower support column (LSC) bodies (cast and non-cast) have been added as expansion links, in addition to the middle axial welds and lower axial welds. The axial weld expansion links are similar to the expansion link in

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[1] Per NSAL 17-1, the 17x17 AXLR style GTs are similar to the 17x17 A and 17x17 AS styles and could be similarly impacted, but there are no Westinghouse units the U.S. with this GT configuration that are presently using ion nitride RCCAs.

MRP-227-A for the equivalent component, the upper and lower core barrel cylinder girth welds, for which the expansion link was the upper and lower core barrel cylinder axial welds.

The addition of the upper core plate is appropriate because it can be susceptible to IE. The primary link for the upper core plate in MRP-227-A was the CRGT lower flange welds, which are not a good predictor for IE because of the relatively low fluence they receive. Alternatively, the LGW is exposed to higher fluence and is a better lead component for IE for the upper core plate. For the LSCs, the change in expansion link is appropriate because the LSCs are susceptible to both IE and IASCC. The CRGT lower flange welds, which was the MPR-227-A primary link for the LSCs, are not susceptible to IASCC, but the LGW is susceptible to IASCC, and thus is a more appropriate lead component for IASCC of the LSCs.

3.1.3.8 Westinghouse Expansion Components

Coverage and Method Changes for Westinghouse Expansion Components

The inspection method and coverage for two Westinghouse expansion components, the Upper Internals Assembly – Item W4.1., "Upper Core Plate and the Lower Internals Assembly," – Item W3.4., "lower support forging or casting," has been changed from EVT-1 examination of 100 percent of accessible surfaces to VT-3 examination of 25 percent of the bottom (non-core side) surfaces. However, both of these items are non-redundant components.

The NRC staff does not generally consider VT-3 examination to be an acceptable examination method for non-redundant components unless these components are highly flaw tolerant. In addition, the examination coverage has been reduced. The NRC staff was concerned that the reduced examination coverage is not sufficient to provide reasonable assurance of component functionality, considering that these are high consequence of failure components.

Therefore, in RAI 14, the NRC staff requested that EPRI a) justify the use of VT-3 examination for these components; b) justify the reduction in examination coverage from 100 percent to 25 percent; c) identify whether it is intended that if the examination of the 25 percent sample of these items reveals indications, the examination coverage will be expanded to include the remaining accessible surfaces of these components.

In the EPRI October 16, 2017, response to RAI 14, both the use of VT-3 and the reduction in coverage from 100 percent to 25 percent are justified based on the design of these components. Both components have numerous holes distributed across the component which would function as crack arrestors. A crack in the ligament between two adjacent holes would not cause loss of function. Many cracked ligaments would be required to cause failure.

VT-3 is acceptable to detect such gross or extensive cracking. The EPRI justification for the use of VT-3 and the reduction in coverage to 25 percent is also based on limited accessibility of the components. Both components have many attached components that would make it difficult to achieve the required camera angles and distance to perform EVT-1 examination, and also block access completely to portions of the components, hence the reduced coverage requirement. EPRI also justified the changes based on the fact that the lower support forging originally had no screened-in degradation mechanisms at all (it was added as an expansion component due to an NRC condition in the SE of MRP-227, Revision 0), and the lower support casting had only loss of fracture toughness due to TE, which has been generically addressed by report PWROG-15032-NP.

EPRI's response also points out that if the expansion to the lower support casting or forging is triggered, an expansion examination of the core barrel lower flange weld will also be triggered. This examination will result in EVT-1 examination of the lower support casting or forging adjacent to the weld since the lower flange weld (LFW) joins the core barrel to the lower support

forging/casting, which is the most likely part of the lower support casting/forging to experience cracking.

With respect to Item c of RAI 14, EPRI indicated that if cracking was found in the upper-core plate (UCP) or lower support forging/casting, the VT-3 examination would be expanded to 100 percent of the accessible areas of the component. EPRI provided a markup of Table 4-6 and Table 5-3 showing this change.

The NRC staff finds the EPRI response to RAI 14 acceptable because EPRI has justified that a VT-3 examination of 25 percent of the component surfaces is sufficient to ensure functionality of the components, given the flaw tolerant nature of the components. In addition, for the lower support forging/casting the region most susceptible to degradation will be examined with EVT-1. Although Table 4-3 lists cracking due to SCC as the effect/mechanism of interest for the lower support forging, the NRC staff considers it unlikely that the forging would experience SCC. RAI 14 is thus resolved.

Therefore, the proposed change in method and coverage for the UCP is acceptable.

3.1.3.9 Westinghouse/CE Primary Components

Inspection Strategy for Westinghouse/CE Core Barrel Welds Sampling

In the initial submittal of MRP-227, Revision 1, EPRI had proposed to reduce the required examination coverage for core barrel primary welds from 100 percent of the accessible weld length (with a minimum of 75 percent of the accessible plus inaccessible weld length required) to a minimum of 25 percent of the circumference. This proposed change was in conjunction with EPRI's proposal to reclassify certain core barrel welds from primary to expansion.

The NRC staff had several concerns regarding the proposed reduction in examination coverage. Therefore, the NRC staff issued RAI 5 requesting additional technical justification of this change. The entire response to RAI 5 is not discussed in detail here because it has been largely overtaken by events, as will be described below. The response to RAI 5 included a functionality evaluation and a discussion of changes to the FMECA for the core barrel welds which has some relevance to the reductions in the minimum required coverage, and also the reclassification of welds from primary to expansion.

In addition, during the 2018 Materials Information Exchange Meeting[2], EPRI MRP reported that, during the spring 2018 outage, a domestic CE plant identified cracks on the outer diameter (OD) surface of the core barrel in the beltline elevation using enhanced visual (EVT-1) examination (Ref. 42). EPRI indicated that one crack-like indication was found in base-metal adjacent to the middle-girth weld, which is a primary component in MRP-227-A, and that several crack-like indications were found in base-metal adjacent to the middle-axial weld, which is an expansion component in MRP-227-A. EPRI stated that industry established a joint EPRI/PWROG Focus Group, with the intent to provide a generic assessment of impact of the core barrel OE to industry. Based on this new information, the staff asked in RAI 29 how this OE would be incorporated into MRP-227, Revision 1.

In its September 28, 2018, response to RAI 29 (Ref. 6), EPRI provided markups of MRP-227, Revision 1, Table 4-2, "CE Plants primary Components," and Table 4-3, "Westinghouse Plants primary Components," that modified the proposed examination coverage of the core barrel welds as shown in Table 1. EPRI also provided markups of MRP-227, Revision 1, Table 4-5, "CE Plants expansion Components," and Table 4-6, "Westinghouse Plants expansion Components,"

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[2] The meeting presentation can be found at Agencywide Document and Access Management System (ADAMS) Package Accession No.: ML18144A252, Ref. 79.

which modified the examination coverage of the associated axial weld expansion components. For the CE primary welds, the table markup includes Note 5 stating that examination coverage requires a minimum of 75% of the length of either the ID or the OD of the weld being examined. For the Westinghouse primary welds, the table markup includes Note 8 stating that examination coverage requires a minimum of 50% of the length of either the ID or the OD of the weld being examined. The staff also notes that the requirement to inspect ¾" of adjacent base metal is consistent with changes proposed in EPRI's response to RAI 20 (Section 3.3).

Table 1 - CE and Westinghouse Core Support Barrel/Core Barrel Primary Welds with Coverage Changes

MRP-227-A Item	Equivalent MRP-227, Revision 1 Item	MRP-227, Revision 1 Coverage Requirement (As modified by RAI 29 response
Core Support Barrel Assembly – Upper (core support Barrel) flange weld	C.5 Core Support Barrel Assembly Upper Flange Weld (UFW)	100 percent of the accessible weld length of one side of the UFW and ¾" of adjacent base metal
Core Support Barrel Assembly – Lower Cylinder Girth Welds	C6. Core Support Barrel Assembly – Middle Girth Weld (MGW)	100 percent of the accessible weld length of the OD of the MGW and ¾" of adjacent base metal
Core Barrel Assembly – Upper core barrel flange weld	W3. Core Barrel Assembly – Upper flange weld (UFW)	100 percent of the accessible weld length of one side of the UFW and ¾" of adjacent base metal
Core Barrel Assembly – Upper and Lower Core Barrel Cylinder Girth Welds	W4. Core Barrel Assembly – Lower Girth Weld (LGW)	100 percent of the accessible weld length of the OD of the LGW and ¾" of adjacent base metal

The response to RAI 5 described a functionality evaluation during a faulted event, with a complete 360° through-wall fracture of a core barrel girth weld. EPRI noted that the secondary core support structure in Westinghouse design plants and the core stops in CE design plants are designed to catch the core barrel and only allow a short drop of the core barrel if it fully fractures. The short drop leaves the upper fuel alignment pins partially engaged, which in turn limits the amount that the top nozzles of the fuel can be offset from the control rod clusters in the upper internals. EPRI stated testing has been conducted that investigated the effect of a full core-drop type accident. Tests were performed at the maximum possible offset and determined that the control rods would still fully insert and that the increased scram times were within acceptable limits. The functionality evaluation also considered the case of a through-wall crack that had propagated around most of the circumference but had left a small remaining ligament.

EPRI also summarized the results of tests of control rod insertability that simulated a full core drop type accident. EPRI stated that core drop testing also tested the effect of significant fuel deflections, in which the center of the fuel assembly was deflected laterally while the top and

bottom were pinned. EPRI determined the effect on scram time was acceptable. EPRI proposed that this testing provides evidence that the small “bend” in the control rod insertion path that could be caused by a tilted core barrel would not impact the ability to insert the control rods for core shutdown. In a letter dated May 17, 2018, EPRI provided additional details of this testing.

EPRI also summarized its functionality evaluation for normal operation. EPRI stated it was likely that a full separation of one of the core barrel girth welds would be detected by the plant loose-parts monitoring system, in-core or ex-core detectors, or some other means. If the condition were not detected the separation would not result in a loss of core support or control rod insertability. However, EPRI stated the effect of the separation on core bypass flow could have an impact on safety.

EPRI also determined a critical crack size for normal operating conditions of around ten percent of the weld circumference. This corresponds to a crack at least several feet in length. EPRI stated this provides further credence to the one-sided inspection of the weld since a long crack is unlikely to form without penetration of the full thickness of the weld.

The EPRI overall conclusions related to functionality in its January 30, 2018, response can be summarized as:

a) Under faulted conditions, the design features included in the reactor vessel internals limit the adverse effects of a complete failure of a core barrel girth weld on the core support or safe shutdown functions of the core barrel.
b) Under normal operating conditions, the critical crack size is at least several feet. This increases the probability of detecting a structurally significant crack.

At a February 15, 2018, public meeting[3], the NRC staff and EPRI discussed the apparent inconsistency of the FMECA documented in MRP-191, Revision 0 (Ref. 67) and Revision 1, with the functionality evaluation results provided in the response to RAI 5. In MRP-191, Revision 0 and Revision 1, both the upper and lower core barrel welds for Westinghouse and CE design RVI were classified as resulting in a high likelihood of core damage if a failure of these welds were to occur. In its letter dated May 17, 2018, EPRI clarified that the high likelihood of core damage in MRP-191, Revision 1, considered both core damage and economic consequence, and that for the core barrel welds, the high likelihood of damage was primarily a result of the likelihood of high economic consequences. EPRI provided revised FMECA results for the core barrel welds that were to be included in MRP-191, Revision 2[4], that separate the safety and economic consequence.

EPRI indicated that in the revised FMECA, the economic consequence is high. Based on a damage likelihood of medium and a failure likelihood of low, the safety category would be “A” based on the FMECA group (which typically corresponds to a “no additional measures” classification), but the FMECA panel conservatively elevated it to “B.” EPRI also noted that the expert panel evaluation was conducted before the core barrel operating experience gained during the spring 2018 outage season, and the recent operating experience at a CE plant could affect the likelihood of degradation, but would not affect the safety consequences. The staff finds that the revised FMECA results are more consistent with the functionality analysis described in the response to RAI 5.

The staff finds the functionality information presented by EPRI increases the staff’s confidence that safe shutdown could be achieved even with a complete or partial girth weld failure, due to

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[3] The meeting presentations can be found at ADAMS Package Accession No.: ML18025B386.
[4] Published November 30, 2018

the design features in the RVI that maintain alignment and support for the core barrel. This would also apply to a full core barrel weld failure during normal operation, although EPRI did not cite the design features for a weld failure during normal operation. However, the NRC staff considers it much less likely that a full core barrel weld failure would occur under normal operating conditions.

In its January 30, 2018, response, EPRI noted that in Westinghouse-designed plants with neutron panels on the outside of the core barrel, only a portion of the weld circumference in the core beltline region is accessible given currently available inspection techniques. The exact percentage of the core barrel circumference covered by the neutron panels varies with plant design. Between 50 and 60 percent of the barrel circumference will be accessible to a visual inspection in a neutron panel plant.

Therefore, the NRC staff understands that the minimum coverage specified in Note 8 to the Table 4-3 markup of 50 percent was chosen by EPRI to accommodate the plants with neutron panels, which are known to have coverage limitations of up to 50 percent. However, the table requires that 100 percent of the accessible weld length be examined, and the NRC staff expects that the Westinghouse plants with thermal shields will achieve much greater than 50 percent coverage, based on data reported in the MRP biannual reports on MRP-227-A RVI examinations, which show thermal shield plants have typically achieved between 75 percent and 90 percent coverage. For the CE core support barrel girth welds, the minimum required coverage of 75 percent is consistent with MRP-227-A, and is therefore acceptable.

The NRC staff finds the proposed revised examination coverage in the response to RAI 29 to be acceptable, because it will require the examination to the maximum extent possible of these core barrel welds, which mitigates the staff concern that structurally significant flaws would be missed by the examination. For the neutron panel plants, the functionality evaluation and revised FMECA results provide reasonable assurance the core barrel will remain functional despite the more limited weld coverage achievable on these welds. The issues documented in RAI 5 and RAI 29 are thus resolved.

<u>Reclassification of Core Barrel Welds</u>

For the Westinghouse core barrel assembly, two welds have been reclassified from primary to expansion in MRP-227, Revision 1. The nomenclature has also been changed for some of the welds and some of the weld items in MRP-227-A have been subdivided in MRP-227, Revision 1. Table 2 below provides the MRP-227-A item name, and the equivalent MRP-227, Revision 1, item name, and shows the breakout into new primary and expansion components of the original component.

Table 2 – Westinghouse Core Barrel Assembly Weld Items Reclassified from Primary to Expansion

Table	MRP-227-A primary Item	MRP-227, Revision 1 primary Item(s)	MRP-227, Revision 1 expansion Item(s)	MRP-227, Revision 1 primary link for expansion Item(s)
4.3	Core Barrel Assembly – Upper and Lower Core Barrel Cylinder Girth Welds	Core Barrel Assembly W4. Lower Girth Weld (LGW) (primary)	Core Barrel Assembly W3.1. Upper Girth Weld (expansion) (UGW)	Core Barrel Assembly W3 Upper Flange Weld (UFW)

Table	MRP-227-A primary Item	MRP-227, Revision 1 primary Item(s)	MRP-227, Revision 1 expansion Item(s)	MRP-227, Revision 1 primary link for expansion Item(s)
4.3	Core Barrel Assembly – Lower Core Barrel Flange Weld	n/a	Core Barrel Assembly W3.3. Lower Flange Weld (LFW)	Core Barrel Assembly W3 Upper Flange Weld (UFW)

For Westinghouse, in MRP-227-A, Table 4-3, “Westinghouse Plants primary Items,” the upper and lower core barrel cylinder girth welds are primary components for cracking due to SCC, IASCC, and fatigue. In MRP-227, Revision 1, Table 4-3, “Westinghouse Plants Primary Items,” the original item has been subdivided into two new items, the LGW and the UGW. Only the LGW is primary in MRP-227, Revision 1, while the UGW has been changed to an expansion item. In addition, the equivalent component to the Core Barrel Assembly – Lower Core Barrel Flange Weld in MRP-227, Revision 1, the Core Barrel Assembly W3.3., “Lower Flange Weld,” has also been reclassified from primary to expansion. The NRC staff notes that the CE item which appears to be analogous to the LFW is C7. "Core Support Barrel Assembly – CSB Flexure Weld (CSBFW)," which remains a primary item in MRP-227, Revision 1.

In addition, per Table 5-3, the expansion to Table 4.6, Core Barrel Assembly W3.2, UAW, would only occur if indications are found in either the UGW or the LFW, which are also expansion items. Thus, the expansion to the UAW would not occur until two refueling cycles had been completed, which could result in as much as four years between the detection of degradation in the primary item until the UAW are examined.

Therefore, in RAI 26, the NRC staff requested that EPRI justify a) reclassifying the UGW from primary to expansion; b) making the UAW a “secondary expansion” to the UGW and LFW; c) reclassifying the LFW from primary to expansion and explain why the LFW classification is not consistent with the analogous CE component, the CSBFW, which is classified as primary.

The October 16, 2017, EPRI response to RAI 26, Item a stated that during preparation for inspections, more details on the typical naming used for the core barrel welds and more precise locations for the welds were found. The information on naming and location provided the basis for the renamed core barrel weld components listed in MRP-227, Revision 1, and the specific aging-related degradation mechanisms assigned to each weld. EPRI stated that this was also an input to the assignment of welds to be primary or expansion components.

EPRI also stated that the upper and lower core barrel cylinder girth welds were revised to provide more detail in MRP-227, Revision 1, resulting in the LGW being subject to SCC, IASCC, IE, and fatigue, and the UGW being subject to SCC.

Further, EPRI stated that in preparing the response to this question, MRP-227, Revision 1 and its base documents were internally reviewed. This internal review found that the degradation mechanisms assigned to the LGW were incorrect. Fatigue should not be assigned as a screened in mechanism. EPRI indicated that based on the original screening and expert panel results, the LGW is only subject to SCC, IASCC, and IE. EPRI indicated the error was present in MRP-227-A and MRP-227, Revision 1. EPRI stated the error will be corrected by deleting “fatigue” from Table 4-3 for inclusion in the final staff-reviewed version of MRP-227, Revision 1, as shown in the markup at the end of the response.

EPRI stated that, provided with this revised information on applicable degradation mechanisms and with the updated nomenclature for each of the welds, the core barrel welds were assigned a more logical structure in MRP-227, Revision 1, using its sampling and lead component strategies. The UGW was made an expansion component to the UFW because both welds have similar low normal operating stresses and were screened in for the same degradation mechanism (SCC). The UFW has the added potential of elevated bending stresses due to the proximity of the upper flange. If the sampling inspection of the UFW as a primary lead component detects evidence of cracking degradation, the UGW will require expansion inspection according to the requirements of MRP-227, Revision 1, to monitor for further occurrence of the degradation mechanism.

Although weld residual stresses are likely the dominant stresses in core barrel welds, and are difficult to predict, the NRC staff finds that EPRI's logic in reassigning the UGW as an expansion component is reasonable, based on the lower operational stresses of the UGW as compared to the UFW (based on a lack of bending stresses in the UGW). The staff also notes that the functionality arguments presented in the response to RAI 5 also support the reclassification of some core barrel girth welds as expansion components, since the response increases the staff's confidence that safe shutdown could be achieved even with a complete failure of a core barrel weld.

The NRC staff therefore finds EPRI's response to RAI 26, Item A to be acceptable.

In response to RAI 26, Item b, EPRI stated that the UAW was assigned as a "secondary expansion" component to the UGW and LFW based on both the likelihood and consequence of degradation in the axial welds as compared to the girth welds. EPRI stated that the UAW is not as severely loaded as a girth weld, so it is less susceptible to cracking degradation mechanisms, and that cracking of the UAW would not result in a loss of core support; therefore, the consequence of failure would be lower than that of a girth weld. EPRI stated that this logic forms the basis for the UAW being made a "secondary expansion" from the UGW and LFW.

The NRC staff finds that the UAW should have lower operating stresses and may be less likely to experience aging degradation than the UGW. Therefore, the staff find's EPRI's logic for making the UAW a secondary expansion is sound and the EPRI response to RAI 26, Item b is acceptable.

In response to RAI 26, Item c, EPRI stated the LFW was made an expansion component to the UFW due to the location experiencing lower operating stresses. Thus, EPRI stated the UFW is expected to lead the LFW in experiencing SCC degradation.

EPRI also stated that the CSBFW in certain CE plant designs is analogous to the LFW in Westinghouse plant designs in terms of the general location on the core barrel; however, the two components are quite different in design. CSBFW is a smaller weld that attaches the CE lower support structure to a flexure on the lower flange of the core barrel. EPRI further stated that this flexure is intended to accommodate relative thermal expansion in the two component assemblies, and that, given the large difference in geometry and design, the two welds are susceptible to different degradation mechanisms. EPRI stated that the CSBFW is subject to both SCC and fatigue, while the LFW is only susceptible to SCC (see part A of this response for a correction to the applicable degradation mechanisms in the LFW). EPRI stated that the differences in design and the differences in applicable degradation mechanisms result in the CSBFW being a primary item while the LFW is reclassified as an expansion component.

The NRC staff finds the EPRI response to RAI 26 Item c to be acceptable because the staff finds the LFW should have lower stresses than the UFW, based on not experiencing the higher bending stresses experienced by the UFW. The NRC staff therefore finds it acceptable for the

LFW to be reclassified as an expansion component. The staff also finds that EPRI provided a reasonable explanation for the CE CSBFW remaining a primary component while the LFW was moved to expansion. The NRC staff also notes that the UGW, UAW, and LFW screened in for SCC, but not for IASCC. SCC has rarely occurred in austenitic stainless steel in PWR reactor coolant other than creviced or stagnant locations, which the core barrel welds are not, due to the water chemistry control in PWR water which minimizes oxidants and contaminants. Therefore, although possible, the NRC staff considers SCC a low-probability mechanism, which also supports these welds being categorized as expansion.

In addition, the NRC staff reviewed the results of EVT-1 examinations of the core barrel primary girth welds in Westinghouse and CE design RVI performed in accordance with MRP-227-A between 2011 and 2018. These examinations are detailed in biennial reports of examination results provided by EPRI (References 83-85). Twenty plants examined at least one of these welds, with 11 Westinghouse and 4 CE plants inspecting all of the required girth welds (4 for Westinghouse, 3 for most CE plants). Average coverage levels in the examinations for the UFW, UGW, and LFW all exceed 95 percent. Five Westinghouse plants have examined only the UFW to date.

Only one plant, a CE unit, found any relevant indications in any of these welds (see discussion earlier in this section). The indications at that plant were found in the middle girth weld and middle axial welds (expansion), which are high-fluence welds susceptible to IASCC. For the welds susceptible to SCC only (UFW, UGW, and LFW), no relevant indications were found. Therefore, in core barrel girth welds which screened in for SCC and not IASCC (e.g., lower fluence girth welds), OE strongly supports that cracking is unlikely in these welds.

Based on the above, it is reasonable to reclassify the UGW and LFW from primary to expansion. RAI 26 is thus resolved.

Additionally, the Core Barrel Assembly – Core Barrel Outlet Nozzle Welds have been deleted as an expansion item from Table 4-6 in MRP-227, Revision 1. In Appendix C to MRP-227, Revision 1, there is a note for this item stating that it has been replaced with three expansion items, the UGW, UAW, and the LFW. The staff finds the deletion of the core barrel outlet nozzle welds to be acceptable because these welds should have relatively low operating stresses. Thus, the welds should be both less susceptible to SCC and highly flaw tolerant. They also have low safety significance because cracking cannot cause displacement of the core barrel and loss of core support.

For CE core support barrel welds and Westinghouse core barrel welds in the "expansion" category, the required examination coverage has been changed in the September, 28, 2018, response to RAI 29 to be consistent with the associated primary welds. For welds with limited access, Note 3 to the markup of Table 4-6 allows 50 percent minimum. Note 4 explains that the MAW and LAW may have access limitations due to neutron panels, and disassembly is not required to access these welds. The staff finds the proposed minimum coverage of 50 percent in Note 3 acceptable because the core barrel girth welds in the expansion category have the same access restrictions as the primary girth welds due to the presence of neutron panels in some plants. The staff finds Note 4 to be acceptable because disassembly of RVI components to permit examination is generally not warranted. In this case, the safety consequence of degradation of the axial welds is lower than that of the girth welds, therefore disassembly is not warranted.

CE & Westinghouse Core Support Columns

In Table 4-2, CE Plants primary Components, Item C8, "Lower Support Structure – Core Support Columns," is a new item that includes both core support columns (for plants with full

height bolted core shroud plates) and core support column welds (for plants with half-height welded core shroud plates). The examination coverage for the core support columns is 25 percent of the column assemblies as visible using a VT-3 examination from above the lower core plate, and for the core support column welds is 100 percent of the accessible surfaces. In MRP-227-A the equivalent item included only the core support column welds, with an examination coverage of 100 percent of the accessible surfaces, for all plants. In MRP-227, Revision 1, there are differences in required examination coverage for the core support column components for the two plant design variations. In addition, the component in Westinghouse-design RVI with the same function is an expansion component whereas the CE core support columns are a primary component.

MRP-227-A has two separate items for Westinghouse Lower Support Assembly - LSC bodies depending on the material (cast or non-cast). In MRP-227, Revision 1, these two items are combined into one in Item W4.4., "Lower Support Assembly – Lower Support Column Bodies (both cast and non-cast)." In addition, the examination method is changed from enhanced visual testing (EVT-1) examination to visual testing (VT-3) examination and the examination coverage is changed from 100 percent of accessible surfaces (for non-cast) or 100 percent of accessible support columns (for cast) to 25 percent of column assemblies as visible using from above the lower core plate.

The NRC staff was concerned that the reduced coverage for the CE core support columns (CSC) and Westinghouse LSC bodies is not sufficient to provide reasonable assurance of component functionality, considering that the LSCs are high consequence of failure components. Also, it is not clear how much information can be gained by a visual inspection from above the core plate.

To resolve these discrepancies, in RAI 9 the NRC staff requested the following information:

a) Justify that the required coverage of 25 percent as visible from above the core plate for Item C8 and W4.4 is sufficient to provide reasonable assurance of functionality.

b) Justify the use of VT-3 examination instead of EVT-1 to detect cracking.

c) Clarify the meaning of "25 percent of column assemblies as visible using a VT-3 examination from above the lower core plate."

d) What expansion of the examination scope to the remaining columns will be conducted if degradation is observed in the 25 percent sample?

e) For CE-design RVI, explain why examination of the core support columns is specified only for plants with full-height bolted shroud plates and not for plants with core shrouds assembled in two vertical sections.

f) Explain why the core support columns are a primary component for CE plants but the component in Westinghouse plants with the same function (lower core support columns) is an expansion component.

In the EPRI January 31, 2018, response to RAI 9 (Ref. 4), Item a indicated the basis for the reduced coverage is the low likelihood of failure and the significant redundancy of the LSCs. EPRI referenced PWROG-14048-P, Revision 1 and the NRC's staff assessment of the report (Ref. 44), as providing the detailed justification. The response indicated that an FMEA was performed for the LSCs followed by a failure tolerance analysis for both the Westinghouse and CE designs. The most limiting functionality case was determined by analysis to be a degraded condition where over 50 percent of the LSCs were failed.

EPRI also provided several factors supporting the assertion that full section cracking of LSCs is unlikely, including: quality controls during fabrication, low ferrite content such that TE or TE plus IE are not expected; low tensile stress; no likely mechanisms for flaw initiation and growth, and sufficient toughness to withstand any secondary cracking in a column that is already cracked through the cross section. In particular, the staff notes that fabrication quality controls included liquid penetrant (PT) examination and radiography, which provides reasonable assurance that before installation the LSCs welds were structurally sound. With respect to RAI 9 Item b, use of VT-3 is adequate because it can detect fully fractured, misaligned, or missing columns, which are the only condition that can affect functionality (e.g., partially cracked columns are still functional).

In response to RAI 9, Item c, EPRI clarified that the intended minimum examination coverage is 25 percent of the overall population of LSCs, both those visible and not visible when viewed from above the lower core plate. EPRI provided a markup of MRP-227, Revision 1, Table 4-6 implementing this clarification.

In response to RAI 9, Item d, EPRI stated that should degradation be observed in the initial inspection population, the examination would expand to include the remainder of the population of the column bodies that are visible through the lower core plate. The response did not indicate the timing for this additional coverage in Section 5. Note 3 of the markup to Tables 4-5 and 4-6 states that:

"…the stated minimum coverage requirement is the minimum if no significant indications are found. However, the Examination Acceptance criteria in Section 5 require that additional coverage must be achieved in the same outage if significant flaws are found. This contingency needs to be considered for inspection planning purposes."

However, as indicated, the criteria in Section 5 do not contain this guidance.

In its May 17, 2018, letter, EPRI proposed to add wording to the "expansion" column of Table 5-3 requiring expansion to the 100 percent of the accessible uninspected column bodies (minimum of 75 percent of the total population) during the same refueling outage the degradation is found. The wording also clarified the relevant conditions for each type of core support column that would trigger the expansion. The NRC staff finds the revised wording is acceptable.

In response to RAI 9, Items e and f, EPRI indicated that the conclusions of PWROG-14048-P provide sufficient technical justification for the core support column welds to be an expansion component for both core shroud designs as discussed in the response to RAI 9, Item a, and that the conclusions of PWROG-14048 that inspection are not necessary to ensure the functionality of the LSC through the PEO are applicable to both CE RVI core shroud designs.

EPRI provided a markup of the MRP-227, Revision 1, Table 4-2 examination requirements for the CE core support columns and core support column welds, which moves the components to Table 4-5, CE Plant Expansion Components, and also implements the clarification discussed in the RAI 9c response. The NRC staff finds the changes to the CSC examinations acceptable.

Additionally, in its May 17, 2018, letter, EPRI proposed changes to the examination coverage descriptions in Tables 4-5 and 4-6 to add guidance that the 25 percent sample of the core support columns and LSCs must be distributed across the lower support structure. Notes were also added to further clarify how this distribution can be done. This change was made to address possible variations in column degradation due to variations in stress and dose from the center to the periphery of the lower support structure, and for consistency with the guidance for CE deep beams (see RAI 10). The staff finds the proposed changes acceptable.

The NRC staff finds EPRI's response to RAI 9 acceptable because it provides a technical justification based on maintaining functionality of the LSC and CSC for the reduction in coverage and the change in examination method for these components, based on a technical report the NRC staff has reviewed and found acceptable. Specifically, in its assessment of PWROG-14048-P, the NRC staff found the report's conclusions to be acceptable, i.e., that full section cracking of lower support columns is extremely unlikely, and that there is significant redundancy in the lower support system (LSS) designs, such that, in the most unlikely event where complete loss of column support does occur at certain locations, the remaining intact structure will sufficiently support limiting loading conditions.

Specifically, greater than 50 percent of the columns can be non-load bearing and the core will remain adequately supported to allow the control rods to be safely inserted. Section 3.5.7 of this SE contains more detail on the staff assessment of PWROG-14048-P, Revision 1. In addition, the EPRI response proposes to move the CE core support columns from the primary category to the expansion category, which is consistent with the categorization of the Westinghouse lower support columns. The NRC staff finds this change acceptable because the CE core support column welds and columns are also addressed by the PWROG-14048-P, Revision 1 report. RAI 9 is thus resolved.

3.1.3.10 CE & Westinghouse Existing Programs Components

In Table 4-8, "CE Plants existing programs Components," a new line item was added for Alignment and Interfacing Components – Core Stabilizing Lugs and Shims. The other information for this line item is identical to that for the Westinghouse clevis insert bolts so this change will be discussed together with the changes for the Westinghouse clevis insert bolts.

There were two changes to Table 4-9, "Westinghouse Plants existing programs Components." The first change was a change in the "Reference" column entry for the "Bottom Mounted Instrumentation System – Flux Thimble Tubes" from NUREG-1801, Revision 1 to IEB 88-09. The NRC staff finds this change acceptable because IEB 88-09 (NRC Bulletin 88-09) is referenced as the basis for GALL AMP XI.M37 for flux thimble tubes.

The second change is the addition of the aging effect of cracking due to SCC for "alignment and interfacing components - clevis insert bolts." Additionally, the "Reference" column has been modified from ASME Code, Section XI, to ASME Code, Section XI as supplemented by TB-14-5. Note 2 to the table states in part that Westinghouse Technical Bulletin TB 14-5 dated August 25, 2014, provides additional information regarding possible visual indication that clevis bolting failure may have occurred, and the note recommends that this information should be reviewed to ensure a heightened awareness of the examiners is applied to this Code inspection. The item description now includes "clevis insert bolts" and "clevis bearing Stellite wear surface".

The NRC staff notes that Westinghouse, Technical Bulletin TB-14-5, Reactor Internals Lower Radial Support Clevis Insert Cap Screw Degradation (Ref. 45), is a Westinghouse proprietary document. TB 14-5 was issued in response to the failure of some lower radial support system (LRSS) clevis-insert bolts at one Westinghouse-design plant in 2010. Westinghouse InfoGram IG-10-1, Reactor Internals Lower Radial Support Clevis Insert Cap Screw Degradation, dated March 31, 2010 (Ref. 46), is a non-proprietary document that provides some detail on the OE related to clevis insert bolt failure.

Clevis insert bolts are fabricated for Alloy X-750, a nickel-based alloy which is susceptible to primary water stress corrosion cracking (PWSCC). The plant that experienced failures detected evidence of the failures of some of the failed bolts visually. Additional detail on the failures in available in PWROG presentation slides, "Industry and NRC Coordination Meeting Materials Programs Technical Exchange: Clevis Insert Bolt Update" (June 2014) (Ref. 47), which

indicates that 29 of a total of 48 clevis insert bolts were failed, and that metallurgical examination confirmed that the failures were caused by PWSCC.

Reference 47 indicated that all designs were inherently safe, and that the impact of clevis insert bolt failures is primarily commercial in nature. Reference 47 also indicates that VT-3 visual examination is an appropriate examination method for clevis insert bolts because it can monitor conditions that affect the functional performance of the LRSS such as excessive or abnormal wear or looseness or dislocation of the clevis insert, and can also reduce the commercial risk related to clevis insert bolt failure by detecting evidence of bolt failure.

Evidence of bolt failure that can be detected via VT-3 visual examination includes wear between the bolt head and lock bar and/or bolt head dislocation. WAAP-8828-P, Revision 0, "Lower Radial Support System (LRSS) Clevis Inserts and Attachment Bolts Design and Safety Function" (Proprietary), March 26, 2014 (Ref. 48), provides additional information on the clevis insert bolts that provides justification that failure of the bolts is not a significant safety issue.

A response to an RAI related to management of clevis insert bolt degradation in the Indian Point, Unit 2 and 3 RVI Program contains information that can be applied to the staff's review generic aging management of clevis insert bolts in the TR. The RAI requested the licensee, Entergy, to justify relying only on the existing ASME Code, Section XI VT-3 visual examination for detection of clevis insert bolt degradation. In its September 27, 2013, response (Ref. 49), Entergy provided a technical justification for the adequacy of the existing inspection requirements. Entergy cited Westinghouse InfoGram IG-10-1, "Reactor Internals Lower Radial Support Clevis Insert Cap Screw Degradation," dated March 31, 2010 (Ref. 46), in support of its response. The key points of the Entergy response are summarized as follows:

a) The main design function of the LRSS that contains the clevis insert bolts (capscrews), is the prevention of tangential or rotational motion of the lower internals assembly while permitting axial displacement and differential radial expansion. These supports are designed to prevent excessive tangential displacement of the lower internals during seismic events and loss-of-coolant accident (LOCA) conditions and also to limit displacements and misalignments in order to avoid overstressing the core barrel and to ensure that the control rods can be freely inserted.

b) The main aging effect of concern is wear due to flow-induced vibration. Failure of capscrews could result in increased wear, which would occur over several cycles (as well as during seismic events and loss of coolant accident (LOCA) conditions) and does not impact the function of the LRSS. This is based on the OE described in the InfoGram (Ref. 46) for the plant that had experienced clevis insert bolt failures.

c) There is a high degree of redundancy in the LRSS. Both IP2 and IP3 have six radial supports spaced at 60-degree intervals around the circumference of the reactor pressure vessel (RPV). Because of the small clearances involved, it is unlikely that complete disengagement of the clevis inserts would occur. If one clevis insert became nonfunctional, the other lower radial supports are capable of resisting all of the internal and external asymmetric loads.

d) Crack detection before bolt failure is not required because of inherent design redundancy.

e) Westinghouse performed an evaluation of the potential for creation of loose parts (and damage from loose parts) caused by clevis insert bolt degradation and concluded that no significant degradation of mechanical components is expected as a result of potential loose parts in the primary system. This is because separated capscrew heads will remain captured in the clevis insert counterbores. Although lock bars experienced wear-

related degradation at the plant with the bolt failures, the potential for damage from loose lock bars is minimal.

f) The visual inspections performed using video cameras during each ten-year interval under ASME Code Section XI are capable of identifying wear or dislodged components of the clevis insert capscrews or dowel pins at any location, if they exist.

g) The Alloy X-750 material used in the IP2 and IP3 clevis insert bolts is not in the most susceptible heat-treatment condition for PWSCC.

Entergy stated in its response that the ASME Code Section XI video camera inspections are capable of identifying wear or dislodged components of the clevis insert capscrews or dowel pins. However, the NRC staff requested additional clarification in a follow-up RAI regarding the ASME Code Section XI inspection of the clevis inserts in order to ensure that the type of degradation documented in Westinghouse InfoGram IG-10-1 would be reliably detected at Indian Point, Units 2 and 3 (IP2 and IP3). In its June 9, 2014, response to the follow up RAI (Ref. 50), Entergy stated that the clevis insert bolts at IP2 and IP3 are inspected as part of the Category B-N-2 Item Number B13.60

Other than the specific heat treatment of the X-750 material, the NRC staff considers the main points of Entergy's response to be generically applicable to all Westinghouse RVI LRSS. Combined with the material presented at the June 2014 materials meeting, the information in Entergy's response confirms that VT-3 examination on a ten-year interval is sufficient to manage clevis insert bolt degradation due to the high degree of redundancy in the LRSS, the low likelihood of creation of significant loose parts, and the fact that it would take several cycles after bolt failure for wear to cause any problems.

Further, referencing the guidance in Westinghouse Technical Bulletin TB 14-5 (Ref. 45) will ensure that nondestructive examination personnel have heightened awareness for the signs of potential clevis insert bolt cracking. The addition of "clevis bearing Stellite wear surface" to this item is also appropriate because it is the examination of the wear surfaces that could detect excessive wear that may be indicative of degradation of the functional performance of the clevis inserts. The NRC staff therefore finds the change to Table 4-9 for the clevis insert bolts acceptable.

CE core stabilizing lugs and shims have the same design functions as Westinghouse clevis insert bolts. The shims are secured to the lugs via bolting that is also susceptible to PWSCC. TB 14-5 also addresses the CE core stabilizing lugs and shims. Therefore, the staff finds the examination method, frequency and coverage for this component to be acceptable.

3.1.4 Changes to Evaluation Methodologies (TR Section 6)

TR Section 6 has been completely revised from MRP-227-A. Section 6 references proprietary topical report BWRVIP-100-A, "BWR [Boiling Water Reactor] Vessel and Internals Project, Updated Assessment of the Fracture Toughness of Irradiated Stainless Steel for BWR Core Shrouds," and MRP-211, "Materials Reliability Program: PWR Internals Age-Related Material Properties, Degradation Mechanisms, Models, and Basis Data - State of Knowledge," (Ref. 82) and sources for irradiated fracture toughness data. Section 6 also refers to IASCC crack growth rate (CGR) models for both PWR and BWR internals in "Models of Irradiation-Assisted Stress Corrosion Cracking of Austenitic Stainless Steels in Light Water Reactor Environments: Volume 1: Disposition Curves Development." (Ref. 76). The staff notes that it has not reviewed or approved this report. Section 6.0 also states that the NRC-accepted report WCAP-17096-NP-A provides approved methods for evaluating examination results that do not meet the examination acceptance criteria in Section 5, and that the WCAP-17096-NP-A methods are to be followed in accordance with the "Needed" requirement of 7.5.

The "Needed" guidance of Section 7.5 specifies that NRC -accepted evaluation methods (ASME Section XI, WCAP-17096-NP or equivalent) must be used for dispositioning examination results that do not meet the examination acceptance criteria in Section 5. The staff finds this needed guidance is sufficient to ensure that NRC-accepted evaluation methods are used, or if not, the NEI 03-08 deviation process will be followed.

The ASME Code is developing a code case based on Reference 76. If this code case is adopted by NRC, it could be used for evaluation of PWR RVI flaws, subject to any NRC conditions on the code case.

3.2 Incorporation of Operating Experience

The NRC staff was interested in how OE informed changes to the inspection and evaluation guidelines in MRP-227, Revision 1. Therefore, RAI 13 requested EPRI to identify 1) any components for which OE has been used to modify or clarify examination coverage requirements based on the actual accessibility achieved during examinations to date, 2) any primary component that was previously considered to be accessible being reclassified as inaccessible due to OE with actual coverage achieved, plus any alternate measures taken for these components, 3) whether expansion links were reevaluated due to reclassification as inaccessible, and 4) whether alternate primary components were selected in such cases.

In its October 16, 2017, letter, EPRI provided two separate responses for RAI 13: one for Westinghouse/CE components and one for B&W components.

3.2.1 B&W Components

The response to RAI 13, Item 1 for B&W components indicated that for one plant, ANO-1, 99 percent coverage was achieved for the CRGT spacer castings because a reactor vessel level monitoring system is installed in one CRGT which blocks access to all but one CRGT spacer casting in that CRGT. Therefore, EPRI proposed a modification to add a note indicating that for ANO-1, acceptable coverage is 680 spacer castings, and acceptable coverage for all other B&W plants is 690 spacer castings.

3.2.2 Westinghouse/CE Components

The EPRI response to Item 1 stated that OE has been applied to modifying and re-classifying the inspection requirements for the core barrel welds. EPRI further stated that during preparation for inspections, more details on the typical naming used for the core barrel welds and more precise locations for the welds were found, and that this information on naming and location provided the basis for the renamed core barrel weld components listed in MRP-227, Revision 1 and the specific aging-related degradation mechanisms assigned to each weld. EPRI stated that this OE was also an input to the assignment of welds to be primary or expansion components.

EPRI stated that the accessibility of the core barrel girth welds behind a thermal shield or neutron pads was a key reason behind the need to justify a lower amount of coverage on the core barrel weld primary items. EPRI further stated that the neutron panel is bolted on the core barrel and inspection of sections of the girth welds that are covered by the neutron panel cannot be completed. EPRI stated that the exact percentage of the core barrel circumference covered by the neutron panels varies with plant design, but that between 50 and 60 percent of the barrel circumference will be accessible to a visual inspection in a neutron panel plant.

EPRI stated that other than the core barrel welds, to date OE has not suggested other changes that are needed to modify or clarify examination coverage requirements for other MRP-227-A

components for Westinghouse plants based on the actual accessibility achieved during the examinations that have been completed.

EPRI also indicated that the CE RVI-design expansion component - ribs and rings - has been determined to be inaccessible for inspection based on a further technical review of drawings, and not based on any OE from prior inspections. EPRI stated that the inspection technique for the Westinghouse RVI- design upper core plate and lower support forging/casting was changed from EVT-1 to VT-3 and coverage was reduced to 25 percent, and that, while there are accessibility concerns with these components, the inspection requirements were not changed as a result of OE.

Summary – RAI 13

In response to RAI 13, Item 1, EPRI explained how coverage requirements were modified and clarified as a result of OE for B&W CRGT spacer castings and Westinghouse and CE core barrel welds. With respect to RAI 13, Items 2, 3, and 4, for both B&W and Westinghouse/CE RVI components, EPRI stated in response to Item 2 that no components have been reclassified as inaccessible based on OE; therefore, no response to Item 3 and 4 was necessary.

The NRC staff finds EPRI's response to RAI 13 acceptable because it explains how OE was used to modify examination coverage requirements in MRP-227, Revision 1. RAI 13 is thus resolved.

3.3 Examination Coverage of Adjacent Base Metal for Welds

For a number of primary and expansion weld items in Tables 4-2, 4-3, 4-5, and 4-6, the revised examination coverage in MRP-227, Revision 1 specifies a percentage of the weld length or circumference "and adjacent base metal" shall be examined. Therefore, in RAI 20, the NRC staff requested EPRI define what extent of the adjacent base metal must be examined (e.g., a certain distance from the weld fusion line or centerline). RAI 20 provided a table listing the specific welds.

In its October 16, 2017 response to RAI 20, EPRI referred to MRP-228, "Materials Reliability Program: Inspection Standard for Pressurized Water Reactor Internals - 2015 Update (MRP-228, Revision 2)," (Ref, 11) for more specific requirements for the adjacent base metal coverage. EPRI proposed the following modification to the text of Section 4.2.2:

> When the adjacent base metal is specified in the inspection coverage requirement, it is intended to include the base metal heat affected zone adjacent to the weld. ~~If no otherwise specified, three quarter inch of base metal coverage may be assumed.~~ The adjacent base metal to be examined is defined in Section 2.3.6.4 of MRP-228 and is referenced in the Section 4 primary and expansion component tables in this document.

The response to RAI 20 also stated that, in order to make this clear, the requirement to inspect ¾" of the adjacent base metal will be added to Tables 4-2 through 4-6 in the final staff-reviewed version of MRP-227, Revision 1 for the welds to which the requirement applies. EPRI provided a table showing the welds where ¾" adjacent base metal inspection is required.

EPRI's response also noted that the base metal examination coverage for Item W2.1 in Table 4-6 was unintentionally omitted, and that this item should have the same base metal examination coverage as item W2 in Table 4-3: 0.25-inch of the base metal adjacent to the lower flange welds on the individual remaining CRGT assemblies. The NRC staff verified that all the welds identified in the table in RAI 20 are included in the table in EPRI's response.

The NRC staff finds that EPRI has adequately addressed RAI 20 by clarifying that the required coverage of base metal adjacent to welds is generally ¾ of an inch on either side of the weld, and that this is a requirement of MRP-227 Revision 1 rather than just a recommendation. The

staff considers that ¾ of an inch is sufficient to encompass the HAZ. Also, EPRI's addition of the ¾ inch examination coverage to Tables 4-2 through 4-6 further ensures that the coverage requirement is clear. RAI 20 is thus resolved.

3.4 Control Rod Drive Mechanism Thermal Sleeve Wear Issue

During the May 2018 Materials Information Exchange Meeting, the Electric Power Research Institute, Materials Reliability Program (EPRI-MRP) and the PWROG Materials Subcommittee made presentations describing recent operating experience with accelerated wear of control rod drive mechanism (CRDM) thermal sleeves (Agencywide Documents Access and Management System (ADAMS) Accession Nos. ML18142A395 and ML18142A457). This wear has the potential to generate loose parts which could jeopardize control rod insertion. Therefore, in RAI 28 the staff requested EPRI to describe how this operating experience will be addressed in MRP-227, Revision 1.

In its September 28, 2018, response to RAI 28, EPRI stated that the CRDM thermal sleeve degradation is not a new issue with Westinghouse PWRs, and that the plants most significantly affected have previously planned or conducted inspections based on Technical Bulletin TB-07-2, Revision 3. However, the safety implications were re-evaluated in light of the recent operating experience as discussed in the Materials Information Exchange Meeting and as communicated in a Part 21 notification. The nuclear steam supply system, original equipment manufacturer has formally notified the industry of the revised evaluation of the degradation by issuing the Westinghouse NSAL-18-1, dated July 9, 2018, and has contacted the potentially affected plants. EPRI stated that this OE was also communicated to PWR plants in April 2018, via EPRI letter MRP 2018-011, dated April 20, 2018. EPRI further stated that the industry is preparing NEI 03-08 interim guidance associated with the NSSS OEM's recommendations in the NSAL-18-1, and that the schedule to issue this interim guidance does not match the current schedule of MRP-227, Revision 1 SE. EPRI stated that if interim guidance is developed, it would be incorporated into MRP-227, Revision 2. EPRI also stated that it is expected that the NSAL/interim guidance will allow industry to gather more extensive baseline inspection information during the next two outage seasons, and thus, NSAL-18-1 and any future interim guidance will better inform industry for generic incorporation during the MRP-227, Revision 2, update effort.

The NRC staff has performed a risk-informed technical evaluation of this issue (Ref. 80). The technical evaluation considered four options, including immediate shutdown of some or all plants. However, the NRC staff determined the potential risk level associated with thermal sleeve wear did not warrant such an action. The option selected was performance of a smart sample through the Operating Experience Smart Sample Program, combined with a generic communication if necessary. This option was chosen because it will allow the NRC staff to gather information on plant-specific issues that would help to determine if the analyses presented in the NSAL and in the technical evaluation are bounding for the plants and have an adequate degree of conservatism. The NRC staff has also issued Information Notice 2018-10 on this topic (Ref. 81). Since EPRI is addressing the CRDM thermal sleeve wear issue through interim guidance and is gathering more inspection data to better inform future guidance on this topic, and the NRC staff, is monitoring this issue through the inspection program, the NRC staff finds that it is acceptable to address this issue in the next revision of MRP-227.

3. 5 Implementation Requirements

MRP-227, Revision 1, Section 7 contains the implementation requirements with respect to the NEI 03-08 protocol, which allows industry issue programs such as the EPRI MRP to designate industry guidance documents or elements as mandatory, needed, or good practice. NEI 03-08 is an industry-controlled program for managing materials integrity in nuclear power plants. As

such, the NRC staff does not endorse the NEI 03-08 document, or the classification of guidance elements under it.

MRP-227, Revision 1 includes Section 7.3, “Reactor Internals Guidelines Inspection Requirement.” In this section, EPRI identifies as “Needed” guidance that “[e]ach commercial U.S. PWR unit shall implement the requirements of Tables 4-1 through 4-9 and Tables 5-1 through 5-3 for the applicable design.”

Section 7.3 in MRP-227, Revision 1, omitted certain information restating portions of the NEI 03-08 deviation process that was previously included in Section 7.3 of MRP-227-A.

Therefore, in RAI 27, the NRC staff requested that EPRI justify the basis for omitting these paragraphs from the scope of Section 7.3 of the MRP-227, Revision 1, report.

EPRI’s October 16, 2017, response to RAI 27 stated that the two aforementioned paragraphs in Section 7.3 of MRP-227-A were removed from the same section of MRP-227, Revision 1 to reduce redundancy between MRP-227-Revision 1 and NEI 03-08. EPRI further stated that MRP-227 contains Mandatory and Needed Requirements under the EPRI MRP Issue Program as stated in Section 7.1 of MRP-227, Revision 1. The "Reactor Internals Guidelines Implementation Requirement" (Section 7.3) contains the “Needed” requirement under NEI 03-08 to implement Tables 4-1 through 4-9 and 5-1 through 5-3 for the applicable plant design. This means that NEI-03-08 requirements apply to the implementation of this document, including Appendix B, Section 8 of NEI-03-08 on Deviations.

EPRI also stated that Appendix B, Section 8 of NEI 03-08 outlines the protocol for utility processing of deviations from Mandatory or Needed requirements. Appendix B, Section 8.1.c. states that “if at any time a utility does not implement any ‘Mandatory’ or ‘Needed" elements of an approved guideline, the utility shall notify the NRC.”

Finally, EPRI stated that the details provided in the paragraphs in question are not considered to be the only possible approaches for dealing with the need for a technical justification. It was not the intention of Section 7.3 of MRP-227 to be prescriptive in how technical justifications should be approached.

Based on EPRI’s response to RAI 27, the staff finds that the removed information is not needed because:

a) It restates the NEI 03-08 deviation protocol, which is not necessary, since Section 7.1 of MRP-227, Revision 1 identifies which elements of the report are “Mandatory” or “Needed” guidance and any licensee that wishes to deviate from this guidance must follow the deviation requirements detailed in NEI 03-08; and

b) It may be interpreted as excessively prescriptive because it suggests options for addressing a deviation from NEI 03-08 “needed” guidance, which may not be the only such options.

RAI 27 is thus resolved.

With respect to timing of the implementation of the needed inspections, Section 7.3 states that for units that have submitted an AMP to the regulator under [referencing] MRP-227-A, and their period of extended operation begins no later than six years from the issuance date of this guideline, that MRP-227-A based program may be implemented as the baseline inspection without deviation from this "Needed" requirement, but the program should also include the requirements contained within the interim guidance letters MRP-2014-006 and MRP-2013-023.

However, subsequent implementation[5] shall be in accordance with the revision of guidelines [MRP-227] in effect at the time.

Section 7.3 further stated that updates of engineering programs for this revision of [MRP-227] for each reactor shall be timely and consistent with the timeframe of required implementation.

Section 7.3 states that if the above exception for submitted AMPs does not apply, a time-period of approximately 36 months (3 years) from the effective date of this revision is considered reasonable for adopting these guidelines. Engineering program updates may be deferred until 24 months (2 years) prior to the beginning of the next inspections governed by these guidelines. The issue date of MRP-227, Revision 1 was October 2015, so plants with initial inspection scheduled to occur through October 2021 may opt to perform these inspections in accordance with MRP-227-A. Subsequent inspections for these plants could potentially be in accordance with a later revision of MRP-227.

3.6 Disposition of Applicant/Licensee Action Items from Staff's SE of MRP-227, Revision 0

In its December 16, 2011, final SE of MRP-227, Revision 0 (Ref. 16), the NRC staff identified eight A/LAIs related to issues that could not be resolved on a generic basis. Since the final SE was issued, the EPRI MRP and PWROG have performed several generic projects with the objective of generically resolving some of the A/LAIs. The following section contains a discussion of the generic work done and experience from plant-specific responses to A/LAIs along with the NRC staff determination as to whether the A/LAI can be modified or eliminated.

3.6.1 A/LAI 1 Applicability of FMECA and Functionality Analysis Assumptions

A/LAI 1 stated:

> Each applicant/licensee is responsible for assessing its plant's design and operating history and demonstrating that the approved version MRP-227 is applicable to the facility. Each applicant/licensee shall refer, in particular, to the assumptions regarding plant design and operating history made in the FMECA and functionality analyses for reactors of their design (i.e., Westinghouse, CE or B&W) which support MRP-227 and describe the process used for determining plant-specific differences in the design of their RVI components or plant operating conditions, which result in different component inspection categories. The applicant/licensee shall submit this evaluation for NRC review and approval as part of its application to implement the approved version of MRP-227.

The NRC staff was concerned that early responses to A/LAI 1 in plant-specific RVI inspection plans were inadequate due to the greater variation in design in the CE and Westinghouse fleets. Supporting analyses and evaluations for MRP-227-A for CE and Westinghouse RVI were performed for "representative plants" rather than using bounding values for parameters such as neutron fluence. Applicants and licensees generally confirmed the three criteria in Section 2.4 of MRP-227-A in response to A/LAI 1, but did not provide further information.

For B&W-design RVI, the NRC staff was less concerned about the bounding nature of the analyses done in support of MRP-227-A, due to the almost identical designs of all seven operating B&W reactors. In addition, A/LAI 1 has been resolved for all six currently operating B&W reactors, as documented in the safety assessments for the RVI inspection programs, or SEs for license renewal (References 52-55).

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[5] TR Section 7.3 states that implementation means performance of examinations of applicable components within the timeframe specified in the applicable tables.

To resolve the generic issue of the information needed from licensees to address A/LAI 1, a series of closed and public meetings were conducted. At these meetings, the NRC staff, Westinghouse, EPRI, and utility representatives discussed regulatory concerns and determined a path for a comprehensive and consistent utility response to demonstrate applicability of MRP-227-A, specifically for Westinghouse and CE-design PWR RVI (Refs. 56-60). A summary of the proprietary meeting presentations and supporting proprietary generic design basis information is contained in Westinghouse proprietary report WCAP-17780-P (Ref. 61). WCAP-17780-P provides background proprietary design information regarding variances in stress, fluence, and temperature in the plants designed by Westinghouse and CE to support NRC staff reviews of utility submittals to demonstrate plant-specific applicability of MRP-227-A.

As a result of the technical discussions with the NRC staff, the basis for a plant to demonstrate compliance with MRP-227-A for originally licensed and uprated conditions was determined to be satisfied with plant-specific responses to the following two questions (as documented in the meeting summaries for the January 22-23 and February 25, 2013, meetings, Refs. 57-58):

> Question 1: Does the plant have non-weld or bolting austenitic stainless steel (SS) components with 20 percent cold work or greater, and, if so, do the affected components have operating stresses greater than 30 ksi? (If both conditions are true, additional components may need to be screened in for stress corrosion cracking, SCC.)
>
> Question 2: Does the plant have atypical fuel design or fuel management that could render the assumptions of MRP-227-A, regarding core loading/core design, non-representative for that plant? (Reference 57 indicated this question covers power uprates as well as other core design and fuel management aspects)

In MRP Letter 2013-025 dated October 14, 2013 (Ref. 62), EPRI provided to licensees a non-proprietary document containing guidance for responding to the two questions above. With respect to Question 1, MRP Letter 2013-025 provides guidance for licensees to assess whether RVI components at their plant, other than those identified in the generic evaluation, have the potential for cold work greater than 20 percent. With respect to Question 2, MRP Letter 2013-025 provides quantitative criteria to allow a licensee to assess whether a particular plant has atypical fuel design or fuel management.

The NRC staff made public its assessment of the guidance in MRP 2013-025 along with the supporting information in WCAP-17780-P on November 7, 2014 (Ref. 63). In the assessment, the staff concluded that if an applicant or licensee demonstrates that its plant(s) comply with the guidance in MRP Letter 2013-025, there is reasonable assurance that IE guidance of MRP-227-A will be applicable to the specific plant(s). The NRC staff further concluded that the guidance in MRP Letter 2013-025 provides an acceptable basis for licensees to prepare responses to generic RAI questions 1 and 2.

The NRC staff also concluded in its assessment that the information provided on evaluation of cold work in WCAP-17780-P provides an adequate technical basis for the guidance in MRP Letter 2013-025 for responding to Question 1. The assessment also stated that the NRC staff concludes that the sensitivity studies of variations in neutron fluence, RVI geometry and temperature documented in WCAP-17780-P, and the information on power uprate effects on fluence and temperature, also documented in WCAP-17780-P, provide an acceptable technical basis for the guidance in MRP Letter 2013-025 for responding to Question 2. The assessment recommended that for responses to Question 2, an applicant/licensee should provide the plant-specific range or value of the numerical parameters (e.g., the average core power density, the heat generation figure of merit "F", and the active fuel to fuel alignment plate distance for CE-design reactors or the active fuel to UCP distance for Westinghouse-design reactors) rather than just stating that the plant complies with the parameter.

In MRP-227, Revision 1, the guidance in MRP Letter 2013-025 is incorporated as Appendix B. Section 2.4 of MRP-227, Revision 1 refers to this guidance for Westinghouse and CE plants.

To generically resolve Question 1 the PWROG developed PWROG-15105-NP, Revision 0, "PA-MSC-1288 PWR RV Internals Cold-Work Assessment," dated April 30, 2016 (Ref. 64), which was submitted to the NRC for information via letter dated June 15, 2016 (Ref. 65). The NRC staff assessment of PWROG-15105-NP, Revision 0, dated April 21, 2017 (Ref. 66), concluded:

a) The majority of austenitic stainless-steel materials were required to be solution annealed, which eliminates the possibility of effects from cold work on the SCC behavior of the materials,

b) Some of the material specifications stipulate limitations on the maximum allowed tensile strength and hardness values, which restricts the possible amount of cold work in the component,

c) No non-fastener, RVI components were subjected to cold work greater than twenty percent in PWR units which makes these components less susceptible to SCC.

d) Material specification and design with respect to the consideration of cold work in CE and Westinghouse non-fastener RVI components did not change over the years of construction of the PWR fleet. Since cold work on these RVI components was adequately controlled during the construction period, it is concluded that non-fastener RVI components from unassessed Westinghouse and CE plants have low cold work and limited susceptibility to SCC.

Based on the above conclusions, the staff finds that a plant-specific response to Question 1 is no longer necessary.

Although a plant-specific response is no longer necessary for Question 1, for Question 2 the NRC staff recommended in its assessment dated November 7, 2014, that plant-specific values of core-design related parameters be documented and provided in the response to Question 2. Based on the above, the NRC staff finds that plant-specific applicability of MRP-227, Revision 1 will be adequately addressed by the criteria of Section 2.4 and Appendix B of MRP-227, Revision 1. The staff recommends that applicants or licensees document this information in their plant-specific RVI program plan, including the plant-specific values, or a plant-specific range of values, of the average core power density, heat generation figure of merit, and applicable dimensional parameter, as described in MRP Letter 2013-025 or Appendix B to MRP-227, Revision 1.

Based on the above, A/LAI 1 is resolved since Section 2.4 and Appendix B of MRP-227, Revision 1 provides adequate guidance to applicants and licensees to ensure the plant-specific applicability of the TR. The resolution of A/LAI 1 in this SE is consistent with the resolution of A/LAI 1 in the NRC's January 29, 2018 SE of Action Items 1 and 7 from MRP-227-A (Ref. 51).

3.6.2 A/LAI 2 PWR Vessel Internal Components within the Scope of License Renewal

A/LAI 2 stated that:

> Consistent with the requirements addressed in 10 CFR 54.4, each applicant/licensee is responsible for identifying which RVI components are within the scope of LR for its facility. Applicants/licensees shall review the information in Tables 4-1 and 4-2 in MRP-189, Revision 1, and Tables 4-4 and 4-5 in MRP-191 and identify whether these tables contain all of the RVI components that are within the scope of LR for their facilities in accordance with 10 CFR 54.4. If the tables do not identify all the RVI components that are within the scope of LR for its facility, the applicant or licensee shall identify the

> missing component(s) and propose any necessary modifications to the program defined in MRP-227, as modified by this SE, when submitting its plant-specific AMP. The AMP shall provide assurance that the effects of aging on the missing component(s) will be managed for the period of extended operation.

The intent of A/LAI 2 in the staff's final SE of MRP-227, Revision 0, was to ensure applicants and licensees identify any plant-specific RVI components that were not addressed by the generic screening, FMECA, functionality, and aging management recommendations of MRP-227-A. This also extends to components that may have the same configuration, but are fabricated from different materials such that they may be subject to different aging mechanisms or effects not identified for the generic component in MRP-227-A.

The NRC staff experience with review of plant-specific RVI Inspection Programs in accordance with MRP-227-A is that most plants have had some plant-specific components, or components with plant-specific materials. However, to date most of these plant-specific components have not required a plant-specific change to the aging management recommendations of MRP-227-A.

MRP-227, Revision 1, Section 2.4 includes the following guidance:

> If major plant-specific differences from the inputs to the FMECA process described in MRP-189 and 191 are identified, then plant owners must determine and document the impact, if any, on the aging management strategy described herein.

Another change to TR Section 2.4 is the additional assumption that the components and material class of each functional component are as listed in the latest revision of MRP-189, or MRP-191, as applicable to the individual plant design. This additional assumption is essentially similar to A/LAI 2 from the NRC staff final SE of MRP-227, Revision 0 (Ref. 16). The staff notes that the latest revisions of these reports which are referenced by MRP-227, Revision 1 are MRP-189, Revision 2 and MRP-191, Revision 1 (Ref. 13). The previous revisions of these reports are MRP-191, Revision 0, and MRP-189, Revision 1.

Based on the above, the TR contains adequate guidance to ensure applicants and licensees will evaluate the impact of any plant-specific components or materials on the aging management guidance for their plants. The NRC staff therefore considers A/LAI 2 resolved.

3.6.3 <u>A/LAI 3 Evaluation of the Adequacy of Plant-Specific existing programs</u>

A/LAI 3 stated:

> Applicants/licensees of CE and Westinghouse are required to perform plant-specific analysis either to justify the acceptability of an applicant's/licensee's existing programs, or to identify changes to the programs that should be implemented to manage the aging of these components for the period of extended operation. The results of this plant-specific analyses and a description of the plant-specific programs being relied on to manage aging of these components shall be submitted as part of the applicant's/licensee's AMP application. The CE and Westinghouse components identified for this type of plant-specific evaluation include: CE thermal shield positioning pins and CE in-core instrumentation thimble tubes (Section 4.3.2 in MRP-227), and Westinghouse guide tube support pins (split pins) (Section 4.3.3 in MRP-227).

A/LAI 3 was included in Ref. 16 because MRP-227-A did not provide adequate guidance for applicants/licensees to document the details of the plant-specific existing programs in plant-specific RVI programs. The NRC staff notes that MRP-227-A did state, with regard to existing plant-specific programs for CE-design RVI. that "the guidance for in-core instrumentation (ICI)

thimble tubes and thermal shield positioning pins is limited to plant-specific recommendations and thus have no generic reference, nor are they included in Table 4-8. The owner should review their specific design, upgrade status, and plant commitments for CE ICI thimble tubes."

With respect to Westinghouse existing plant-specific programs, MRP-227-A stated that:

> The guidance for guide tube support pins (split pins) is limited to plant-specific recommendations and thus have no generic reference. Subsequent performance monitoring should follow the supplier recommendations. Thus, they are not included in Table 4-9. The owner should review their specific design, upgrade status, and asset management plans for guide tube support pins (split pins).

For Westinghouse split pins, similar guidance is included in MRP-227, Revision 1, to that in MRP-227-A. The revised guidance states "Additionally, in Westinghouse–design plants, the originally installed alloy X750 guide tube support pins (split pins) have been typically replaced with components with improved designs and less susceptible materials. The plant owner should review their specific design, upgrade status, and asset management plans for guide tube support pins (split pins). Thus, the guide tube support pins (split pins) are not included in Table 4-9." However, the guidance is not sufficient because it does not specify that an applicant or licensee must include the specifics of the AMP for split pins in its plant-specific RVI program. Also, the revised wording appears to imply that aging management is only necessary for Alloy X-750 split pins.

Therefore, in RAI 15 the staff requested EPRI:

a) Clarify if type 316 stainless steel split pins require a plant-specific AMP. Modify the wording of section 4.4 of MRP-227, Revision 1, as necessary.

b) Include a requirement in MRP-227, Revision 1, that the specific AMP for split pins be documented in the plant-specific RVI program, including the replacement and/or inspection schedule, replacement material, examination method and coverage, technical basis for the replacement schedule or the remaining life of the split pins (if already replaced), and technical basis for the inspection schedule or lack of inspections.

In its October 16, 2017, response to RAI 15, EPRI clarified that Type 316 stainless steel split pins are "no additional measures" components, therefore do not require a plant-specific AMP. EPRI also provide a markup of Section 4.4 that includes this clarification and also states that licensees with Alloy X750 split pins must perform a plant-specific evaluation to determine appropriate aging management, and document this evaluation in their RVI Inspection Program, until the licensee replaces the split pins with Type 316 stainless steel split pins.

The NRC staff finds EPRI's response to RAI 15 acceptable because it clarifies that Type 316 split pins do not require an aging management program, and because more definitive language will be added to MRP-227, Revision 1, requiring documentation of the plant-specific AMP for Alloy X-750 split pins. The staff concern in RAI 15 is thus resolved. With the inclusion of this language in MRP-227, Revision 1, for split pins, A/LAI 3 is no longer needed for Westinghouse-design plants.

In MRP-227, Revision 1, the guidance, with regard to CE existing plant-specific items, has been eliminated. The NRC staff reviewed the status of the resolution of A/LAI 3 for the CE units implementing MRP-227-A. The staff has received submittals for 8 of the 10 operating CE units for review of the plant-specific RVI programs. The NRC staff has approved the RVI programs for all 8 of these units. A/LAI 3 was either not applicable or successfully resolved for all 8 of these units. For the two remaining CE units, the licensee does not have a commitment to submit the RVI program for NRC staff review.

Therefore, A/LAI 3 is resolved or expected to be resolved as part of implementation of RVI programs based on MRP-227-A for all CE units that have commitments to submit RVI programs for staff review and approval. A/LAI 3 can be eliminated for CE-design RVI in the accepted version MRP-227, Revision 1.

Based on the above, A/LAI 3 is resolved and does not need to be included in this SE.

3.6.4 A/LAI 4 B&W Core Support Structure Upper Flange Stress Relief

A/LAI 4 stated that:

The B&W applicants/licensees shall confirm that the core support structure upper flange weld was stress relieved during the original fabrication of the Reactor Pressure Vessel in order to confirm the applicability of MRP-227, as approved by the NRC, to their facility If the upper flange weld has not been stress-relieved, then this component shall be inspected as a "primary" inspection category component If necessary, the examination methods and frequency for non-stress relieved B&W core support structure upper flange welds shall be consistent with the recommendations in MRP-227, as approved by the NRC, for the Westinghouse and CE upper core support barrel welds. The examination coverage for this B&W flange weld shall conform to the staff's-imposed criteria as described in Sections 3.3.1 and 4.3.1 of this SE The applicant's/licensee's resolution of this plant-specific action item shall be submitted to the NRC for review and approval.

Licensees for all six operating B&W plants responded to A/LAI 4 in their plant-specific RVI program submittals, which have all been accepted by the NRC staff (Refs. 52-55) verifying that stress relief had been performed on the core support structure upper flange welds for each plant. Therefore, A/LAI 4 has been acceptably resolved and is eliminated for MRP-227, Revision 1.

3.6.5 A/LAI 5 Application of Physical Measurements as part of IE Guidelines for CE, and Westinghouse RVI Components

A/LAI 5 stated:

> Applicants/licensees shall identify plant-specific acceptance criteria to be applied when performing the physical measurements required by the NRC-approved version of MRP-227 for loss of compressibility for Westinghouse hold down springs, and for distortion in the gap between the top and bottom core shroud segments in CE units with core barrel shrouds assembled in two vertical sections The applicant/licensee shall include its proposed acceptance criteria and an explanation of how the proposed acceptance criteria are consistent with the plants' licensing basis and the need to maintain the functionality of the component being inspected under all licensing basis conditions of operation during the period of extended operation as part of their submittal to apply the approved version of MRP-227.

The hold-down spring in Westinghouse-design RVI prevents flow-induced vibration of the lower internals. If the hold-down spring loses too much preload, wear of the mating surfaces (RPV flange, upper internals top support plate, and lower internals core barrel flange) could result. The NRC staff understands that the required compressive force for the hold down spring is best determined just before the outage during which the physical measurement is performed, since it can vary with the fuel design used and mass flow rates.

The NRC staff issued RAIs to some applicants/licensees of Westinghouse-design plants asking for a description of the methodology for determining the hold-down spring height acceptance criteria. The methodology described in all these responses assumes a linear decrease in spring height from the preservice measurement to the required spring height at end of life (60 years)

with the acceptance criteria at any point in time in between determined by interpolation. Some applicants also indicated that they plan to replace the hold-down spring with Type 403 stainless steel material that is not susceptible to stress relaxation, thus does not require physical measurements.

Table 6-5 of MRP-191, Revision 0, indicates the hold-down spring is FMECA group 1 (low consequence) and that degradation of the hold-down spring could cause significant economic impact but does not threaten safe shutdown or lead to a breach of fuel cladding. Due to the consistent, conservative methodology used to determine the hold-down spring height acceptance criteria, and the low-safety consequence of degradation of the hold-down spring, the NRC staff determined that it does not need to review the plant-specific acceptance criteria for the hold-down spring. Therefore, A/LAI 5 can be eliminated for the Westinghouse hold-down spring.

With respect to the gap between core shroud segments in CE plants, any measureable separation between the upper and lower core shroud segments is considered a relevant condition per MRP-227, Revision 1, Table 4-2. The specified examination method is visual VT-1 examination which is capable of resolving very small gaps based on the requirement to resolve a character height of 0.044 inches. Any detection of such a gap would trigger an engineering evaluation using the NRC-accepted methodology of WCAP-17096-NP-A. WCAP-17096-NP-A, Revision 2 provides acceptable guidance for determination of whether detected gaps between CE core shroud segments are acceptable. The CE core shroud assembly (welded) is item CE-ID: 5 in WCAP-17096-NP-A, Revision 2, and the guidance is found on pages C-30-C31.

Further, plant-specific RVI programs have been submitted for eight of ten operating CE units and the NRC staff has accepted the programs for all eight. A/LAI 5 was not applicable for five units of the eight CE units for which RVI programs have been submitted to NRC. For the remaining three, A/LAI 5 has been resolved. Therefore A/LAI 5 has been resolved or determined to be inapplicable for eight CE units as part of implementation of MRP-227-A. The licensee of the remaining two units does not have a commitment to submit its RVI program for review.

Based on the above, A/LAI 5 can be eliminated for the gap between core shroud segments for CE plants, because detection of any gap would trigger an evaluation using the NRC-accepted methodology in WCAP-17096-NP-A, Revision 2, or the latest NRC accepted version of WCAP-17096. WCAP-17096-NP-A, Revision 2 provides acceptable guidance for determining if detected gaps are acceptable. In addition, A/LAI 5 has been resolved for all applicable CE units during implementation of MRP-227-A. Therefore, it is not necessary to include this A/LAI in this SE.

Based on the above, A/LAI 5 is eliminated for all PWR designs in MRP-227, Revision 1.

3.6.6 A/LAI 6 Evaluation of Inaccessible B&W Components

A/LAI 6 stated:

> MRP-227 does not propose to inspect the following inaccessible components: the B&W core barrel cylinders (including vertical and circumferential seam welds), B&W former plates, B&W external baffle-to-baffle bolts and their locking devices, B&W core barrel-to-former bolts and their locking devices, and B&W core barrel assembly internal baffle-to-baffle bolts. The MRP also identified that although the B&W core barrel assembly internal baffle-to-baffle bolts are accessible, the bolts are non-inspectable using currently available examination techniques.

> Applicants/licensees shall justify the acceptability of these components for continued operation through the period of extended operation by performing an evaluation, or by proposing a scheduled replacement of the components. As part of their application to implement the approved version of MRP-227, applicants/licensees shall provide their justification for the continued operability of each of the inaccessible components and, if necessary, provide their plan for the replacement of the components for NRC review and approval.

The NRC staff notes that all the components listed under A/LAI 6 are expansion components in both MRP-227-A and MRP-227, Revision 1, and continue to be classified as inaccessible in MRP-227, Revision 1. The staff notes that WCAP-17096-NP-A, Revision 2 provides guidance for performing the engineering evaluations of these inaccessible components. Further, Condition 1 of the NRC staff SE of WCAP-17096-NP, Revision 2, requires the licensee to submit the detailed analyses, replacement schedule, or justification for some alternative process, for the three inaccessible expansion component items within one year of the inspection of the linked primary component items for NRC staff to determine whether review is needed, if the inspection results indicate aging triggering the expansion criteria in Table 5-1 of MRP-227-A. The SE for WCAP-17096-NP, Revision 2, identifies the three inaccessible expansion component items as:

a) The core barrel cylinder and welds

b) The former plates; and

c) The core barrel-to former bolts and the internal and external B-B bolts.

These items are consistent with those listed in A/LAI 6. The NRC staff SE of WCAP-17096-NP, Revision 2 only requires these evaluations be submitted to the NRC if degradation is detected in the linked primary items. However, this is consistent with the philosophy of MRP-227, since expansion items are only considered to be susceptible to degradation if the same type of degradation is first detected in the linked primary item. In such cases, the examination of expansion components is typically not required until one to three refueling outages after detection of aging in the linked primary component.

Submittal of the evaluation of the inaccessible components within one year is sufficiently timely to allow NRC staff review of these evaluations before any action would typically be required for the expansion items. Therefore, since the A/LAI 6 requirement to submit the evaluation of the inaccessible components is addressed by Condition 1 of the NRC staff SE of WCAP-17096-NP, Revision 2, A/LAI 6 is eliminated for MRP-227, Revision 1.

3.6.7 A/LAI 7 Plant-Specific Evaluation of CASS Materials

A/LAI 7 stated:

> The applicants/licensees of B&W, CE, and Westinghouse reactors are required to develop plant-specific analyses to be applied for their facilities to demonstrate that B&W IMI guide tube assembly spiders and CRGT spacer castings, CE lower support columns, and Westinghouse lower support column bodies will maintain their functionality during the period of extended operation or for additional RVI components that may be fabricated from CASS, martensitic stainless steel or precipitation hardened stainless steel materials. These analyses shall also consider the possible loss of fracture toughness in these components due to thermal and irradiation embrittlement, and may also need to consider limitations on accessibility for inspection and the resolution/sensitivity of the inspection techniques. The requirement may not apply to components that were previously evaluated as not requiring aging management during

development of MRP-227. That is, the requirement would apply to components fabricated from susceptible materials for which an individual licensee has determined aging management is required, for example during their review performed in accordance with Applicant/Licensee Action Item 2. The plant-specific analysis shall be consistent with the plant's licensing basis and the need to maintain the functionality of the components being evaluated under all licensing basis conditions of operation. The applicant/licensee shall include the plant-specific analysis as part of their submittal to apply the approved version of MRP-227.

A/LAI 7 was included in the NRC staff SE of MRP-227, Revision 0, due to concerns about TE as well as the potential for a synergistic effect of TE and IE for CASS RVI components that are susceptible to TE and receive sufficient fluence to be subject to IE. Since the NRC staff final SE of MRP-227, Revision 0, the PWROG issued several technical reports intended to generically address aspects of A/LAI 7.

By letter dated March 13, 2015 (Ref. 68), the PWROG submitted PWROG-14048-P to the NRC for information only. This report contains a generic methodology for evaluating the functionality of Westinghouse and CE LSCs. The NRC staff assessment of PWROG-14048-P, Revision 0 (Ref. 70) concluded:

a) The analyses for assessing the failure likelihood of the LSCs in Section 5 of the report utilized bounding inputs, such as high membrane stresses and saturated values of material-fracture toughness, to demonstrate that the likelihood of full-section failure of LSCs is low.

b) The failure tolerance evaluation of the LSCs to demonstrate structural redundancy in the LSS as discussed in Section 6 of the report presents a reasonable approach for addressing structural redundancy in the LSS. However, due to plant-specific differences, each plant must consider its specific design parameters when establishing the tilt and deflection criteria and the assumed spread or cluster of failed LSCs.

c) Consideration of buckling needs to be included for generic acceptance of the redundancy analysis presented in Section 6. In addition, when evaluating scenarios where an assumed spread or cluster of LSCs has lost its support function, plant-specific evaluations that consider the potential for buckling and for changes in the modal characteristics of the LSS need to be included.

The PWROG revised PWRG-14048-P with the intention of demonstrating that the LSC functionality analysis is generically applicable to all Westinghouse and CE units. The staff assessment of PWROG-14048-P, Revision 1 (Ref. 72) concluded that:

a) The flaw tolerance analyses of the four LSC designs representing participating plants demonstrate that the likelihood of full-section failure of LSCs is low.

b) The approach in evaluating structural redundancy of the LSS assembly of the four LSC designs representing participating plants is reasonable, except for the aspect of buckling discussed in the next paragraph; that the four LSC designs adequately addresses the range of plant-specific geometric parameters, loading conditions, and acceptance criteria of participating plants; and that structural redundancy evaluation adequately included the effect of clusters of failed LSCs.

c) The discussion of LSC buckling in the redundancy analysis adequately addressed buckling of an LSC subject only to a compressive axial load but did not address the

effect of eccentric[6] loading in LSC buckling; and the discussion of the LSS assembly dynamic response described in the report adequately shows there is little change in the dynamic response of the LSS assembly due to failed LSCs.

In “Final Safety Evaluation of Action Items 1 and 7 from Topical Report MRP-227-A, ‘Materials Reliability Program: Pressurized Water Reactor Internals Inspection and Evaluations Guideline,’" (Ref. 51), the NRC staff determined the issue of bucking due to eccentric loading does not represent a safety issue, based on the following. The faulted condition analyzed in PWROG-14048, Revision 1 is very conservative and an unlikely condition since LOCA and seismic events are assumed to occur at the same time.

Furthermore, the NRC staff determined that the flaw tolerance evaluation in the report demonstrated that the likelihood of full-section failure of the LSCs is low. This means that the likelihood of having an LSC configuration with broken LSCs is low. Since high eccentric loads occur under faulted loads for cases with broken LSCs, the likelihood of having LSCs subject to high eccentric loads is even lower. Therefore, the NRC staff determined there is reasonable assurance that the LSCs, and the LSS, would remain functional under design-basis conditions. The staff therefore concluded in that no plant-specific analysis is necessary to address the effect of eccentric loads on LSC buckling.

PWROG-15032-NP describes the statistical analysis of a large number of heats of CASS material used in U.S. PWRs in order to determine statistical upper bounds on the ferrite content. The report demonstrates that there is a high probability that all low-molybdenum CASS (Type CF3 or CF8) used in U.S. RVI is below the screening criterion for TE of 20 percent delta ferrite.

The NRC staff assessment of PWROG-15032-NP (Ref. 73) concluded that the report can be used by applicants or licensees to estimate the delta ferrite content for Type CF8 (static or centrifugally cast) and static-cast Type CF3M CASS components without the need to obtain the plant-specific CMTRs. These estimated ferrite values may then be used to screen the CASS material for TE. The staff assessment also stated that for CASS components subject to neutron fluences greater than $1x10^{17}$ n/cm^2, additional adjustments for the effects of irradiation must be applied to the methodology used to estimate toughness.

However, the NRC staff has subsequently revised its position on screening criteria for loss of fracture toughness for CASS RVI components exposed to neutron fluence. The technical basis for these criteria is documented in Appendix A to the NRC staff SE of the BWRVIP-234, "Thermal Aging and Neutron Embrittlement Evaluation of Cast Austenitic Stainless Steel for BWR Internals," (Ref. 74). The new screening criteria allow loss of fracture toughness to be ruled out for statically cast low-molybdenum CASS RVI components with ferrite content less than or equal to 20 percent, and centrifugally cast components with ferrite content less than or equal to 25 percent, that will experience neutron exposures of 0.00015 to 1 displacement per atom (dpa) ($1x10^{17}$ n/cm^2 to $6.7x10^{20}$ n/cm^2). Below $1x10^{17}$ n/cm^2, CASS components need to be screened for TE only using the existing screening criteria in the May 19, 2000, NRC staff letter from Christopher Grimes (NRC) to Douglas J. Walters (NEI).

Based on the revised screening criteria for loss of fracture toughness, a loss of functionality of low-molybdenum CASS components due to loss of fracture toughness only needs to be considered for components that will have neutron exposures greater than 1 dpa (fluence greater than $6.7x10^{20}$ n/cm^2). Based on PWROG-15032-NP, TE can be screened out for all low-molybdenum CASS components.

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[6] An eccentric load is defined as a compressive, axial, off center load. An eccentric load could cause buckling at a lower value than would be required for buckling if the load was center-aligned.

A/LAI 7 was only intended to apply to RVI components for which additional aging management activities beyond ASME Code, Section XI ISI was specified in MRP-227-A; e.g., components other than “no additional measures” components.

The only generic CE CASS RVI component that is not a “no additional measures” component is the CSCs in some plants. PWROG-14048-P, Revision 1 provides a generic methodology for evaluating functionality of CSCs in CE plants. Therefore, for all CE CASS RVI components that are not “no additional measures components,” loss of fracture toughness has been adequately addressed.

Westinghouse RVI generic components that may be CASS and are not “no additional measures” components consist of the LSCs, Lower Support Casting, and the CRGT lower flanges. PWROG-14048-P, Revision 1 provides a generic methodology for evaluation of loss of fracture toughness for Westinghouse LSCs. The lower support casting is exposed to low neutron fluence, thus TE is the only mechanism for loss of fracture toughness. TE is generically eliminated as a concern for the lower support casting via report PWROG-15032-NP. The CRGT lower flange (welds) are inspected for cracking as a “primary” component, thus aging of this component is adequately managed. Therefore, for all Westinghouse CASS RVI components that are not “no additional measures” components, loss of fracture toughness has been adequately addressed.

B&W RVI generic components fabricated from CASS or precipitation-hardened stainless steel that are not “no additional measures” components include the CRGT spacer castings, the IMI guide tube spiders, and the vent valve retaining rings (15-5 PH stainless steel). A generic approach has not been developed to resolve A/LAI 7 for B&W-design RVI. With respect to B&W components, the NRC staff determined that A/LAI 7 is resolved for three licensees (five units) based on functionality evaluations submitted by these licensees, as documented in staff SEs for Oconee, Units 1, 2, and 3, ANO-1, and TMI-1 (Refs. 52, 54, and 75). The licensee for Davis-Besse, the one remaining B&W unit, made a LR commitment to submit its A/LAI 7 evaluation at least one year prior to the scheduled MRP-227-A examinations of the applicable components (Ref. 69). Because that licensee for the one remaining B&W unit made a commitment to submit its plant-specific evaluation, the staff does not consider it necessary to retain A/LAI 7 in this SE.

Based on the above, A/LAI 7 has been acceptably resolved for all generic CASS components in plants with CE-design and Westinghouse-design RVI, and plant-specific responses to A/LAI 7 are no longer necessary for applicants or licensees of these plants submitting RVI Programs in accordance with MRP-227-A. A/LAI 7 has also been resolved for five of six operating B&W reactors. Licensees of plants with B&W-design RVI should submit plant-specific A/LAI 7 evaluations in accordance with existing commitments, if they have yet to do so.

It is possible that plant-specific CASS components could be identified in applicant or licensee in accordance with the guidance of TR Section 2.4. If these components are not classified as “no additional measures” components, then appropriate aging management activities must be identified for the components per TR Section 2.4. The ferrite content of these components may be estimated using the methodology and information in PWROG-15032-NP. Screening of these components for IE and TE may be done according to the criteria found in Appendix B to the Final BWRVIP-234 SE.

The resolution of A/LAI 7 in this SE is consistent with the resolution of A/LAI 7 in the NRC’s January 29, 2018, safety evaluation of Action Items 1 and 7 from MRP-227-A (Ref. 51).

3.6.8 A/LAI 8 Submittal of Information for Staff Review and Approval

A/LAI 8 states that:

> Applicants/licensees shall make a submittal for NRC review and approval to credit their implementation of MRP-227, as amended by this SE, as an AMP for the RVI components at their facility. This submittal shall include the information identified in Section 3.5.1 of [the staff's final SE of MRP-227, Revision 0]. The items detailed in Section 3.5.1 are 1) an RVI AMP, 2) RVI inspection plan, 3) FSAR supplement, 4) any necessary TS changes needed to manage aging of RVI, and 5) any TLAAs related to RVI. For licensees that already have a renewed license, A/LAI 8 only requires the first two items. The fifth item also states in part that for those CUF analyses that are TLAAs, the applicant may use the PWR Vessel Internals Program as the basis for accepting these CUF analyses 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 in the components during the period of extended operation, and that the periodicity of the inspections of these components shall be justified to be adequate to resolve the TLAA. The fifth item also included wording requiring that the fatigue TLAAs for RVI address the effects of the reactor water environment.

The NRC staff reviewed A/LAI 8 and determined that it is not necessary to retain this action item for MRP-227, Revision 1. Only Items 1 and 2 above were required for all applicants and licensees. These address submission of an AMP and RVI inspection plan.

The NRC staff determined that Items 1 and 2 are acceptably addressed by applicant/licensee commitments, which typically require the submission of the licensee's RVI program for NRC review and approval no later than two years prior to the PEO, for licensees that received renewed licenses prior to the completion of the development of the industry RVI program in MRP-227, Revision 0. Items 3, 4, and 5 are only required for LR applicants that submitted LRAs after the issuance of the staff's final SE of MRP-227, Revision 0. Items 3, 4, and 5 are all addressed by the regulation at 10 CFR Part 54 and are therefore redundant.

The additional information in Item 5 regarding the periodicity of inspections for fatigue would be addressed through the NRC staff review of the disposition of the fatigue TLAA for RVI. The statement in Item 5 that "To satisfy the evaluation 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" adds an additional technical requirement to the review of the fatigue TLAA for RVI.

The RVI of most PWRs were not designed to the requirements of ASME Code, Section III, subsection NG since the code of record for most PWRs predates the inclusion of Subsection NG in the Code. Therefore, this imposes an additional requirement not in the CLB for most plants. Further, in the Standard Review Plan for License Renewal, NUREG-1800, Revision 2, the guidance is clear that the effects of the reactor water environment only need to be addressed for ASME Code, Section III, Class 1 reactor coolant pressure boundary components.

Therefore, the recommendation in Item 5 to address the effects of the environment for RVI fatigue CUFs is not consistent with NRC staff guidance for license renewal. Based on the above, A/LAI 8 can be eliminated for MRP-227, Revision 1.

3.7 Referencing of MRP-227, Revision 1 in LR or SLR Applications

Both NUREG-1801, Revision 2and NUREG-2191 include a recommended AMP for PWR internals in Section XI.M16A. AMP XI.M16A in both the GALL Report and GALL-SLR Report is based on MRP-227-A. The GALL-SLR guidance specifies that applicants must perform a gap analysis to identify any changes to the guidance of MRP-227-A that are needed to manage aging out to 80 years, since MRP-227-A was developed with the assumption that end of life is 60 years.

Licensees that update the RVI AMP to follow the NRC-accepted version of MRP-227, Revision 1, will need to identify an exception to the GALL, until the NRC staff revises its guidance in the GALL and GALL-SLR to reference the NRC-accepted version of MRP-227, Revision 1 as the basis for the PWR Internals AMP.

4.0 APPLICANT/LICENSEE ACTION ITEMS

Applicants or licensees that find degradation of BFBs shall comply with the following:

A/LAI 1

If the table in MRP 2017-009 indicates that the subsequent inspection interval is not to exceed 6 years (e.g., downflow plants with ≥ 3 percent BFBs with indications or clustering, or upflow plants with ≥ 5 percent of BFBs with indications or clustering), the plant-specific evaluation to determine a subsequent inspection interval shall be submitted to the NRC for information within one year following the outage in which the degradation was found. Any evaluation to lengthen the determined inspection interval or to exceed the maximum inspection interval recommended in MRP-2017-009 shall be submitted to the NRC for information at least one year prior to the end of the current applicable interval for BFB subsequent examination.

5.0 CONCLUSIONS

The NRC staff finds that MRP-227, Revision 1, as modified by this SE, provides an acceptable means for managing aging of PWR reactor vessel internals. MRP-227, Revision 1, as modified by this SE, is acceptable for referencing in LR applications to satisfy the requirements of 10 CFR 54.21(a)(3) for demonstrating that the effects of aging on the RVI components within the scope of MRP-221, Revision 1, will be acceptably managed. Applicants or licensees implementing MRP-227, Revision 1, shall address A/LAI 1 as described in Section 4.0 of this SE. A licensee that desires to implement an RVI inspection program in accordance with MRP-227, Revision 1, as modified by this SE, to fulfill a LR commitment, must submit its RVI inspection program in accordance with its existing LR commitment.

The NRC staff finds MRP-227, Revision 1, as modified by this SE and subject to the A/LAI detailed in Section 4.0 of this SE, provides an acceptable baseline or starting point for an AMP for SLR subject to a gap analysis as described in the SRP-SLR Section 3.1.2.2.9 and GALL-SLR, AMP XI.M16A. An exception to GALL-SLR AMP XI.16A must be identified in such cases.

6.0 REFERENCES

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3. Letter from Mike Hoehn II, Ameren Missouri, MRP Integration Committee Chairman and Brian Burgos, EPRI, MRP Program Manager Subject: Responses To NRC Request For Additional Information For Electric Power Research Institute Topical Report MRP-227, Revision 1, “Materials Reliability Program: Pressurized Water Reactor Internals Inspection And Evaluations Guideline” (CAC NO. MF7740), October 16, 2017, (ADAMS Accession No. ML17305A056)

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53. Three Mile Island Nuclear Station, Unit 1 - Letter and Non-Proprietary Safety Evaluation of Reactor Vessel Internals Inspection Plan (TAC No. MF1459), December 19, 2014 (ADAMS Accession No. ML14297A411)

54. Arkansas Nuclear One, Unit 1 - Redacted Version - Staff Assessment Regarding Program Plan for Aging Management for Reactor Vessel Internals (CAC No. MF4201) February 13, 2017 (ADAMS Accession No. ML16333A338)

55. NUREG-2193, Supplement 1, "Safety Evaluation Report Related to the License Renewal of Davis-Besse Nuclear Power Station. Docket Number 50-346, FirstEnergy Nuclear Operating Company," April 30, 2016 (ADAMS Accession No. ML16104A350)

56. U. S. Nuclear Regulatory Commission Letter, “Summary of November 28, 2012, Category II Public Meeting with the Electric Power Research Institute and Industry Representatives,” January 29, 2013 (ADAMS Accession No. ML13009A066)

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61. Westinghouse Report, WCAP-17780-P, Revision 0, “Reactor Internals Aging Management MRP-227-A Applicability for Combustion Engineering and Westinghouse Pressurized Water Reactor Designs,” June 2013

62. Electric Power Research Institute (EPRI), MRP-227-A Applicability Template Guideline, October 14, 2013 (ADAMS Accession No. ML13322A454) (MRP Letter 2013-025)
63. Evaluation of WCAP-17780-P, "Reactor Internals Aging Management MRP-227-A Applicability for Combustion Engineering and Westinghouse Pressurized Water Reactor Designs," and MRP-227-A, Applicability Guidelines for CE, November 7, 2014 (ADAMS Accession No. ML14309A484)
64. PWROG-15105-NP "PA-MSC-1288 PWR RV Internals Cold-Work Assessment, Materials Committee," April 30, 2016 (ADAMS Accession No. ML17075A195)
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72. Staff Assessment of Report PWROG-14048-P, Revision 1, "Functionality Analysis: Lower Support Columns," August 30, 2017 (ADAMS Accession No. ML17242A003)
73. Staff Assessment of the Pressurized Water Reactor Owner's Group Report PWROG-15032-NP, Revision 0, "PA-MSC-1288 Statistical Assessment of PWR RIV Internals CASS Materials" (TAC No. MF7223), August 25, 2016 (ADAMS Accession No. ML16222A254)
74. Final BWRVIP-234 Safety Evaluation and Transmittal Letter, June 22, 2016 (ADAMS Accession No. ML16096A002)
75. Three Mile Island Nuclear Station, Unit 1 - Non-Proprietary, Staff Assessment of Action Item 7 Regarding Inspection Plan for Reactor Internals (EPID L-2016-LLL-0002), July 12, 2018 (ADAMS Accession No. ML18159A108)
76. Models of Irradiation-Assisted Stress Corrosion Cracking of Austenitic Stainless Steels in Light Water Reactor Environments; Volume 1: Disposition Curves Developments, EPRI, Palo Alto, CA: 2014. 3002003103
77. PWROG-17071-NP, "WCAP-17096-NP-A Interim Guidance," March 31, 2018 (ADAMS Accession No. ML18204A166)
78. PWR Owners Group Submittal of PWROG-17071-NP, "WCAP-17096-NP-A Interim Guidance" to the NRC for Information Only (PA-MSC-1567), July 12, 2018 (ADAMS Accession No. ML18204A165)

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80. CRDM Thermal Sleeve LIC-504 - Rev 2, September 27, 2018 (ADAMS Accession No. ML18249A107)
81. NRC Information Notice 2018-10: "Thermal Sleeve Flange Wear Leads to Stuck Control Rod at Foreign Nuclear Plant," August 29, 2018 (ADAMS Accession No. ML18214A710)
82. EPRI Report 1015013, "Materials Reliability Program: PWR Internals Age-Related Material Properties, Degradation Mechanisms, Models, and Basis Data - State of Knowledge (MRP-211)," December 31, 2007 (ADAMS Accession No. ML093020614)
83. MRP 2014-009, Biennial Report of Recent MRP-227-A Reactor Internals Inspection Results (Project 694), May 12, 2014 (ADAMS Accession Nos. ML14135A383, ML14135A384, and ML14135A385)
84. MRP 2016-008, Biennial Report of MRP-227-A Reactor Internals Inspection Results, May 18, 2016 (ADAMS Accession No. ML16144A789)
85. EPRI - 2018 Biennial Report of Recent MRP-227-A Reactor Internals Inspection Results, July 19, 2018 (ADAMS Accession No. ML18204A161)

Attachment: Comment Resolution Table

Principal Contributor: Jeff Poehler, NRR

Date:

EPRI Comments on Draft Safety Evaluation and NRC Staff Disposition					
No.	**Page**	**Line**	**Comment**	**Potential Revision**	**NRC Staff Disposition**
1	All	All	Industry has confirmed that there is no proprietary information in the report.	No redactions are required	N/A
2a	21-22	21: 42-44 22: 1-10	There are plants other than Catawba, Unit 2 that should be listed as an exception for the CRGT guide card inspection timing (Seabrook is an example that has completed initial guide card inspections in accordance with FME video sampling inspection in Table 5-14 of WCAP-17451-P during the refueling outage in spring 2017 and documented in EPRI letter MRP 2018-025, dated 7/19/2018 [ML18204A161]). These plants were addressed through the bullets about "Plants with FME videos analyzed" in PWROG letter OG-18-76 (Reference 41 in the SER): "• Plants with FME Videos Analyzed in Table 5-14 – According to Table 5-14 Schedule"	Potential change to sentence starting on Page 21, Line 43: "The interim guidance accelerates the baseline examination schedule for the plants with 17x17 A, 17x17 AS, and 17x17 AXLR[1] guide tubes (e.g., those addressed in the 10 CFR Part 21 notification) to either 2018 or 2020, except for Catawba, Unit 2, which already performed baseline guide card wear measurements in 2016, and plants with foreign material exclusion videos analyzed in Table 5-14 of WCAP-17451-P, which can perform the baseline guide card wear measurements according to the Table 5-14 schedule."	Incorporated as suggested.
2b-1	20	29	Title of WCAP-17451-P should be "Westinghouse Domestic Fleet Operational ***Projections***"	"Westinghouse Domestic Fleet Operational ***Projections***"	Incorporated

2b-2	55	14	Title of WCAP-17451-P should be "Westinghouse Domestic Fleet Operational ***<u>Projections</u>***"	"Westinghouse Domestic Fleet Operational ***<u>Projections</u>***"	Incorporated
2c	28	8	The last sentence of this paragraph states: "The NRC staff notes that the CE item equivalent to the LFW is C7. "Core Support Barrel Assembly – CSB Flexure Weld (CSBFW)," which remains a primary item in MRP-227, Revision 1." This is technically incorrect. The CSBFW is not equivalent to the LFW. This was specifically addressed in the response to RAI 26.		Replaced "equivalent" with "appears to be analogous".
3	50	38	Regarding the proposed A/LAI #1 associated with the baffle-former-bolt (BFB) plant-specific analysis, as discussed during the 9/12/2018 public meeting, the industry team considers that the most appropriate location for this A/LAI (or Condition) is in the SE for topical report WCAP-17096-NP Revision 3. The industry has incorporated the NRC's request regarding utility submittals of plant-specific assessments for BFB reinspection periods into the update to WCAP-17096-NP Revision 3.		The staff will retain A/LAI #1 associated with the BFB plant-specific analysis. The industry's suggestion for the staff to include the A/LA in the SE for WCAP-17096-NP Revision 3 would likely not be implemented for several years, since WCAP-17096-NP, Rev. 3 has not been submitted yet to NRC. Not including the A/LAI in this SE would mean that there would be a period of several years with no NRC guidance directing the plant-specific analyses be submitted to NRC.
4	numerous	various	General comment: numerous places do not specifically state that an RAI has been resolved (as worded on Page 5 Line 18). Examples follow (not necessarily all inclusive).		A statement has been added after the discussion of the staff evaluation of each RAI stating "RAI X is thus resolved." The change was

					made for several other RAIs in addition to those specifically listed in the following comments.
4a	7	9	No summary statement is given as to resolution of the RAI.	Add: RAI 4 is thus resolved.	Incorporated.
	9	4	No summary statement is given as to resolution of the RAI.	Add: RAI 11 is thus resolved.	Incorporated.
	9	33	No summary statement is given as to resolution of the RAI.	Add: RAI 17 is thus resolved.	Incorporated.
	11	14	No summary statement is given as to resolution of the RAI.	Add: RAI 21 is thus resolved.	Incorporated.
	15	6	No summary statement is given as to resolution of the RAI.	Add: RAI 16 is thus resolved.	Incorporated.
	15	48	No summary statement is given as to resolution of the RAI.	Add: RAI 12 is thus resolved.	Incorporated.
	16	17	No summary statement is given as to resolution of the RAI.	Add: RAI 24 is thus resolved.	Incorporated.
	19	28	No summary statement is given as to resolution of the RAI.	Add: RAI 8 is thus resolved.	Incorporated.

Acknowledgments

The following organizations, under contract to the Electric Power Research Institute (EPRI), prepared this report:

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EPRI, Palo Alto, CA: 2019.
3002017168.

Report Summary

The Materials Reliability Program (MRP) developed inspection and evaluation (I&E) guidelines for managing the long-term aging of reactor vessel internal components of pressurized water reactors (PWRs). Specifically, the guidelines are applicable to PWR internals structural components; they do not address fuel assemblies, reactivity control assemblies, or welded attachments to the reactor vessel. Revision 1 of these guidelines provides updates based on the Nuclear Regulatory Commission (NRC) Safety Evaluation Report for MRP-227, Revision 0, as well as operating experience and new knowledge gained from various materials testing, modeling and research projects. Revision 1-A of these guidelines incorporates changes from the NRC Safety Evaluation Report for MRP-227, Revision 1.

Background

Demonstrating that the effects of aging degradation in PWR internals are adequately managed is essential for maintaining a healthy fleet and assuring continued functionality of PWR internals. As a work product of the MRP, these I&E guidelines are intended to support that demonstration, with requirements for various inspections to detect effects of aging degradation. The program to develop and maintain these guidelines is organized around a framework and strategy for managing effects of aging in PWR internals, and is supported by a substantial database of material data and evaluation results. The goal of this development was primarily to support license renewal. The requirements contained in this document are applicable to Babcock & Wilcox (B&W), Combustion Engineering (CE), and Westinghouse Nuclear Steam Supply System (NSSS) "Generation II" PWR designs constructed and initially operating in the United States prior to 2007.

Objectives

- To provide generic I&E guidelines for each PWR design for use by individual plant owners in developing engineering programs to manage aging in PWR internals.
- To evaluate PWR internals for all three PWR designs currently operating in the United States, and make recommendations for aging management actions specific to each component.
- To support the industry in preparing PWR internals aging management programs (AMPs) to satisfy license renewal commitments.

Approach

An experienced team consisting of utility and nuclear steam supply system (NSSS) vendors and Electric Power Research Institute (EPRI) experts, representing a broad spectrum of reactor design, operations, and materials expertise, worked on the project. The team

developed screening criteria with susceptibility levels for the eight postulated aging mechanisms relevant to PWR internals and their effects. Initial component screening and categorization was completed using susceptibility levels and FMECA (failure modes, effects, and criticality analysis) to identify the relative ranking of components. The team also completed engineering evaluations and safety assessments of degradation for components and assemblies of components. Aging management strategy development, combining results of the engineering evaluations and safety assessments with component accessibility, operating experience, existing evaluations, and prior examination results, was completed to determine the appropriate aging management methodology, baseline examination timing, and the need for and the timing of subsequent inspections.

Results

One "mandatory," and four "needed" Nuclear Energy Institute (NEI) 03-08 implementation requirements have been developed. These requirements provide the framework and details for individual utility engineering programs for managing aging in reactor internal components, and the development of AMPs to support license renewal.

Applications, Value, and Use

The guidelines are based on a broad set of assumptions about plant operation, which, with some exceptions, encompass the range of current plant conditions for the U.S. fleet of PWRs. The aging management strategies reports (MRP-231 and MRP-232) provide the basis for these guidelines. The functional evaluations that support the guidelines were based on representative configurations and operational histories, which were generally conservative, but not necessarily bounding in every parameter. These assumptions are a conservative representation of U.S. PWR operating plants, all of which implemented low-leakage core-loading patterns early in their operating life. The recommendations are thus applicable to all U.S. PWR operating plants as of May 2007 for the three designs identified. These guidelines are also considered applicable to plants that have replaced components or component assemblies; however, alternatives can be technically justified. The Inspection Standard for PWR internals (MRP-228) is the companion document to these I&E guidelines. It provides examination requirement standards for components listed in the guidelines.

Keywords

Pressurized water reactor, Reactor internals, Inspection guidelines, Aging management, License renewal, Materials Reliability Program

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Product Type: Technical Report

Product Title: Materials Reliability Program: Pressurized Water Reactor Internals Inspection and Evaluation Guidelines (MRP-227, Revision 1-A)

PRIMARY AUDIENCE: Program engineers at pressurized water reactor (PWR) utilities

SECONDARY AUDIENCE: Inservice inspection engineers

KEY RESEARCH QUESTION

The current generation of PWR plants is approaching the end of their respective licensing periods, and some plants have already entered their periods of extended operation. The nuclear power industry in the United States developed Inspection and Evaluation (I&E) guidelines for managing the long-term aging of PWR internals components of pressurized water reactors. Revision 1 of these guidelines provides updates based on the Nuclear Regulatory Commission (NRC) Safety Evaluation Report for MRP-227, Revision 0, as well as operating experience and new knowledge gained from various materials testing, modeling, and research projects. Revision 1-A of these guidelines incorporates changes from the NRC Safety Evaluation Report for MRP-227, Revision 1. As a living program, the I&E guidelines will continue to be periodically updated.

RESEARCH OVERVIEW

An experienced team consisting of utility and nuclear steam supply system (NSSS) vendors and Electric Power Research Institute (EPRI) experts, representing a broad spectrum of reactor design, operations, and materials expertise, worked on the project. The team developed screening criteria with susceptibility levels for the eight postulated aging mechanisms relevant to PWR internals and their effects. Initial component screening and categorization was completed using susceptibility levels and FMECA (failure modes, effects, and criticality analysis) to identify the relative ranking of components. The team also completed engineering evaluations and safety assessments of degradation for components and assemblies of components. Aging management strategy development, combining results of the engineering evaluations and safety assessments with component accessibility, operating experience, existing evaluations, and prior examination results, was completed to determine the appropriate aging management methodology, baseline examination timing, and the need for and the timing of subsequent inspections.

KEY FINDINGS

- Generic I&E guidelines are provided for each PWR design for use by individual plant owners in developing engineering programs to manage aging in PWR internals.
- Recommendations for aging management actions specific to each component are provided for the PWR internals of all three PWR designs currently operating in the United States.
- Support is provided for preparing PWR internals aging management programs (AMPs) to satisfy license renewal commitments.
- One “mandatory,” and four “needed” Nuclear Energy Institute (NEI) 03-08 implementation requirements have been developed.

WHY THIS MATTERS

Demonstration that the effects of aging degradation in PWR internals are adequately managed is essential for maintaining a healthy fleet and assuring continued functionality of the PWR internals. As a work product of the EPRI Materials Reliability Program (MRP), these I&E guidelines are intended to support that demonstration, with requirements for various inspections to detect the effects of aging degradation. These guidelines are provided to individual plant owners for use in preparing and executing their PWR internals aging management programs.

HOW TO APPLY RESULTS

The guidelines are based on a broad set of assumptions about plant operation, which, with some exceptions, encompass the range of current plant conditions for the U.S. fleet of PWRs. The aging management strategies reports (MRP-231 and MRP-232) provide the basis for these guidelines. The functional evaluations that support the guidelines were based on representative configurations and operational histories, which were generally conservative, but not necessarily bounding in every parameter. These assumptions are a conservative representation of U.S. PWR operating plants, all of which implemented low-leakage core-loading patterns early in their operating life. The recommendations are thus applicable to all U.S. PWR operating plants as of May 2007 for the three designs identified. These guidelines are also considered applicable to plants that have replaced components or component assemblies; however, alternatives can be technically justified. The Inspection Standard for PWR internals (MRP-228) is the companion document to these I&E guidelines. It provides examination requirement standards for components listed in the guidelines.

LEARNING AND ENGAGEMENT OPPORTUNITIES

- MRP Assessment and Inspection Technical Advisory Committees

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PROGRAM: Pressurized Water Reactor Materials Reliability Program (MRP), P41.01.04

IMPLEMENTATION CATEGORY: Technical Basis Report, Reference

List of Acronyms

A/LAI	Applicant/Licensee Action Item
AMP	Aging Management Program
ANO-1	Arkansas Nuclear One Unit 1
ASME	American Society of Mechanical Engineers
B&PV	Boiler & Pressure Vessel
B&W	Babcock & Wilcox
BB	Baffle-to-Baffle
BMI	Bottom Mounted Instrumentation
BWR	Boiling Water Reactor
BWRVIP	Boiling Water Reactor Vessel & Internals Project
CAP	Corrective Action Program
CASS	Cast Austenitic Stainless Steel
CB	Core Barrel
CE	Combustion Engineering
CEA	Control Element Assembly
CFR	Code of Federal Regulations
CGR	Crack Growth Rate
CRGT	Control Rod Guide Tube
CSA	Core Support Assembly
CSS	Core Support Shield
DB	Davis-Besse
E	Expansion, I&E Guidelines Component Group
EFPY	Effective Full Power Years
EPFM	Elastic-Plastic Fracture Mechanics
EPRI	Electric Power Research Institute
EPU	Extended Power Uprate
ET	Electromagnetic Testing (Eddy Current)
EVT	Enhanced Visual Testing (a Visual NDE Method that includes EVT-1)
FB	Baffle-to-Former
FD	Flow Distributor
FME	Foreign Material Exclusion

FMECA	Failure Mode, Effects, and Criticality Analysis
GALL	Generic Aging Lessons Learned
HAZ	Heat-Affected Zone
HWC	Hydrogen Water Chemistry
I&E	Inspection and Evaluation
IASCC	Irradiation-Assisted Stress Corrosion Cracking
IC	Irradiation-Enhanced Creep
ICI	In-Core Instrumentation
ID	Inner Diameter
IE	Irradiation Embrittlement
IGSCC	Intergranular SCC
IMI	Incore Monitoring Instrumentation
IP	Issue Program
IPA	Integrated Plant Assessment
ISI	Inservice Inspection
ISR	Irradiation-Enhanced Stress Relaxation
ITG	Issue Task Group
JOBB	Joint Owners Baffle Bolt
LCB	Lower Core Barrel
LCP	Lower Core Plate
LEFM	Linear Elastic Fracture Mechanics
LTS	Lower Thermal Shield
LOCA	Loss-of-Coolant-Accident
MRP	Materials Reliability Program
N	No Additional Measures, I&E Guidelines Component Group
NDE	Non-Destructive Examination
NEI	Nuclear Energy Institute
NRC	U.S. Nuclear Regulatory Commission
NSSS	Nuclear Steam Supply System
OBE	Operating Basis Earthquake
OD	Outer Diameter
OEM	Original Equipment Manufacturer
ONS	Oconee Nuclear Station (ONS-1, ONS-2, and ONS-3)
P	Primary, I&E Guidelines Component Group
PH	Precipitation-Hardenable (via Heat Treatment)
PMMP	Preventive Maintenance Management Program
PWR	Pressurized Water Reactor
PWROG	Pressurized Water Reactor Owners Group
PWSCC	Primary Water SCC
RCCA	Rod Cluster Control Assembly
RCS	Reactor Coolant System

RI	Reactor Internals
RV	Reactor Vessel
SCC	Stress Corrosion Cracking
SE	Safety Evaluation
S/N	Serial Number
SS	Stainless Steel
SSE	Safe Shutdown Earthquake
SSHT	Surveillance Specimen Holder Tube
TE	Thermal Aging Embrittlement
TLAA	Time-Limited Aging Analysis
TMI-1	Three Mile Island Unit 1
UCB	Upper Core Barrel
UCP	Upper Core Plate
U.S.	United States
USP	Upper Support Plate
UT	Ultrasonic Testing (a Volumetric NDE Method)
UTS	Upper Thermal Shield
VS	Void Swelling
VT	Visual Testing (a Visual NDE Method that includes VT-1 and VT-3)
X	Existing, I&E Guidelines Component Group
XL	Extra-Long Westinghouse Fuel

Record of Revision

0	Original Issue
0, A	This revision incorporates the Topical Report Conditions resulting from the NRC Safety Evaluation Review and responses to associated Requests for Additional Information (see Appendix B). It also contains minor editorial corrections identified since the original issuance of the guidelines.
1	This revision incorporates recent changes as agreed upon by EPRI MRP Reactor Internals Core Team Writers and provides updates based on the NRC Safety Evaluation Review for MRP-227, Revision 0, as well as operating experience and new knowledge gained from various materials testing, modeling, and research projects.
1, A	This revision incorporates the Topical Report Conditions resulting from the NRC Safety Evaluation Review of MRP-227 Revision 1 and responses to associated Requests for Additional Information (see Appendix D). It also contains minor editorial corrections identified since the original issuance of the guidelines. See Appendix C for detailed changes to the guidelines.

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Section 1: Executive Summary

Demonstration that the effects of aging degradation in pressurized water reactor (PWR) internals are adequately managed is essential for maintaining a healthy fleet and assuring continued functionality of the PWR internals. As a work product of the Electric Power Research Institute (EPRI) Materials Reliability Program (MRP), these Inspection & Evaluation (I&E) guidelines are intended to support that demonstration, with requirements for inspection to detect the effects of aging degradation. These guidelines are provided to individual plant owners for use in preparing and executing their PWR internals aging management programs. These guidelines contain Mandatory and Needed requirements that must be implemented per the Materials Initiative [1]. Section 7 describes the requirements of the guidelines, including an implementation schedule. The requirements contained in this document are applicable to Babcock & Wilcox (B&W), Combustion Engineering (CE), and Westinghouse Nuclear Steam Supply System (NSSS) "Generation II" PWR designs constructed and initially operating in the United States prior to 2007.

These guidelines are not intended to reduce, alter, or otherwise affect current American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel (B&PV) Code Section XI [2] or plant-specific licensing inservice inspection requirements. Where ASME Code Section XI examinations are credited for aging management as "Existing Programs," these guidelines provide the specificity considered necessary to ensure that the examinations meet the intent for which they are credited.

The program to develop and maintain these guidelines has been underway since the mid-2000s. It is organized around a framework and strategy for managing the effects of aging in PWR internals, and is supported by a substantial database of material data and evaluation results. The key sequential steps included the following:

- Development of screening criteria, with susceptibility levels for the eight postulated aging mechanisms relevant to PWR internals and their effects;
- Initial component screening and categorization, using the susceptibility levels to identify the relative susceptibility of the components;

- Engineering evaluations and safety assessments of degradation for components and assemblies of components;
- Aging management strategy development combining the results of engineering evaluations and safety assessments with component accessibility, operating experience, existing evaluations, and prior examination results to determine the appropriate aging management methodology, baseline examination timing, and the need for and the timing of subsequent inspections.

Through this process, the PWR internals for each PWR design were assigned to one of the following four groups: Primary, Expansion, Existing Programs, and No Additional Measures components. Definitions and recommendations for aging management actions specific to each group are provided in Sections 3 and 4.

The aging management elements needed for Primary and Expansion components were selected from existing, well-proven visual, surface, and volumetric examination methodologies that have been subject to widespread, relevant application. Each component in the Primary and Expansion groups was then assessed in terms of the degradation effect (e.g., cracking caused by particular mechanisms, loss of material caused by wear), appropriate examination methodology for detection of that effect, accessibility of that component for the examination method selected, and industry experience with those examinations. The Inspection Standard for PWR internals (MRP-228) [3] is the companion document to these I&E guidelines and provides the examination requirement standards for the components listed herein.

The Primary component requirements are listed in Tables 4-1, 4-2, and 4-3 of Section 4 for the B&W, CE, and Westinghouse designs, respectively. The Expansion component requirements are listed in Tables 4-4, 4-5, and 4-6 for the B&W, CE, and Westinghouse designs, respectively. These tables provide the assembly/sub-assembly/component description, the relevant degradation effect and associated degradation mechanism, any link between a Primary component and a related Expansion component, the examination method, examination coverage, initial examination baseline schedule requirements, and examination frequency.

The Existing Programs component requirements are listed in Tables 4-8 and 4-9 for the CE and Westinghouse designs, respectively. There are no Existing Programs components for B&W plants. These tables and the supporting text identify the components and the references to the existing programs.

Tables are not provided for the No Additional Measures components. This group of components has been determined to need no additional aging management. However, for those components in the No Additional Measures group that are classified as core support structures in plant-specific documentation, the inservice inspection requirements of ASME Code Section XI, Subsection IWB, Examination Category B-N-3 [2] must continue

to be met, unless specific relief is granted as allowed by Title 10 Part 50.55a [4] of the Code of Federal Regulations (10 CFR 50.55a) or plant-specific licensing documentation. See MRP-231 [13] and MRP-232 [14] for additional information on No Additional Measures components.

The examination acceptance criteria and the expansion criteria for the primary/expansion links are described in Tables 5-1, 5-2, and 5-3 for the B&W, CE, and Westinghouse designs, respectively. These examination acceptance criteria include visual examination relevant conditions that require disposition by additional examinations, engineering evaluation, or repair/replacement.

Section 6 briefly discusses evaluation methods for the disposition of findings from inspections completed in accordance with Section 4. *Comprehensive guidance on condition evaluation is under the scope of the PWROG technical report WCAP-17096-NP-A* [26].

Section 2: Introduction

2.1 Background

This document provides inspection and evaluation (I&E) guidelines for use by the industry to develop engineering programs to manage aging in PWR internals. It is also intended to support a utility in developing its PWR internals aging management program. This program is an NEI-03-08 industry initiative and an element of an aging management program (AMP) for PWR internals. The program applies to current operation, upratings, and 60 year license renewal NRC commitments.

Thus for the purpose of this document the following are defined and differentiated:

Engineering program – an administratively controlled and ongoing engineering activity (controlled by an owner-specific procedure or equivalent document) that implements regulatory requirements, industry recommendations, plant efficiency and safety improvements, industry operating experience (OE), or self-imposed utility elected requirements for the management of PWR internals continued reliability (asset management) and safety-related functionality.

Aging Management Program – an administratively controlled document prepared to demonstrate that the elements are in place, which satisfies the license-renewal regulatory guidance found in NUREG-1801. It would typically credit in part the engineering program described above for satisfying this requirement.

The goal of these I&E guidelines is to ensure the long-term safety, integrity, and reliability of PWR internals using proven and familiar methods for inspection, monitoring, surveillance, and reporting. The purpose of this Revision 1-A is to:

- Maintain guidance as a 'living program' with consideration of recent inspection results and research developments
- Incorporate considerations related to Generic Responses for Applicant/Licensee Action Items (A/LAIs) and conditions to the maximum extent practical
- Incorporate new research results / testing information, as well as Operating Experience

- Incorporate lessons learned from initial implementation of MRP-227-A (and Rev.0) at utilities
- Incorporate information garnered from plant-specific questions from NRC staff reviewers
- Adopt WCAP-17096-NP as an approved evaluation methodology

These guidelines are organized around a framework and strategy [5] for managing the effects of aging in PWR internals, together with a substantial database of material data and supporting results (e.g., see [6]). The key steps in the framework and strategy process are shown in the flowchart of Figure 2-1.

Based upon the framework and strategy, and on the accumulated data, three important precursor elements to these guidelines were then developed:

- screening criteria, considering chemical composition, neutron fluence exposure, temperature history, and representative stress levels, for determining the relative susceptibility of PWR internals to the eight postulated aging mechanisms [7] – stress corrosion cracking (SCC), irradiation-assisted stress corrosion cracking (IASCC), wear, fatigue, thermal aging embrittlement (TE), irradiation embrittlement (IE), irradiation-enhanced stress relaxation and creep (ISR/IC), and void swelling (VS);
- categorization of PWR internals, based on the screening criteria and the likelihood and severity of safety and economic consequences, into categories that range from those components for which these issues are insignificant (Category A) to those components that are potentially moderately significant (Category B) to those components that are potentially significantly affected (Category C) [8, 9, and 10]; and
- Engineering evaluations and safety assessments of components and assemblies of components based on representative plant designs using irradiated and aged material properties to determine the effects of the degradation mechanisms on functionality [11 and 12].

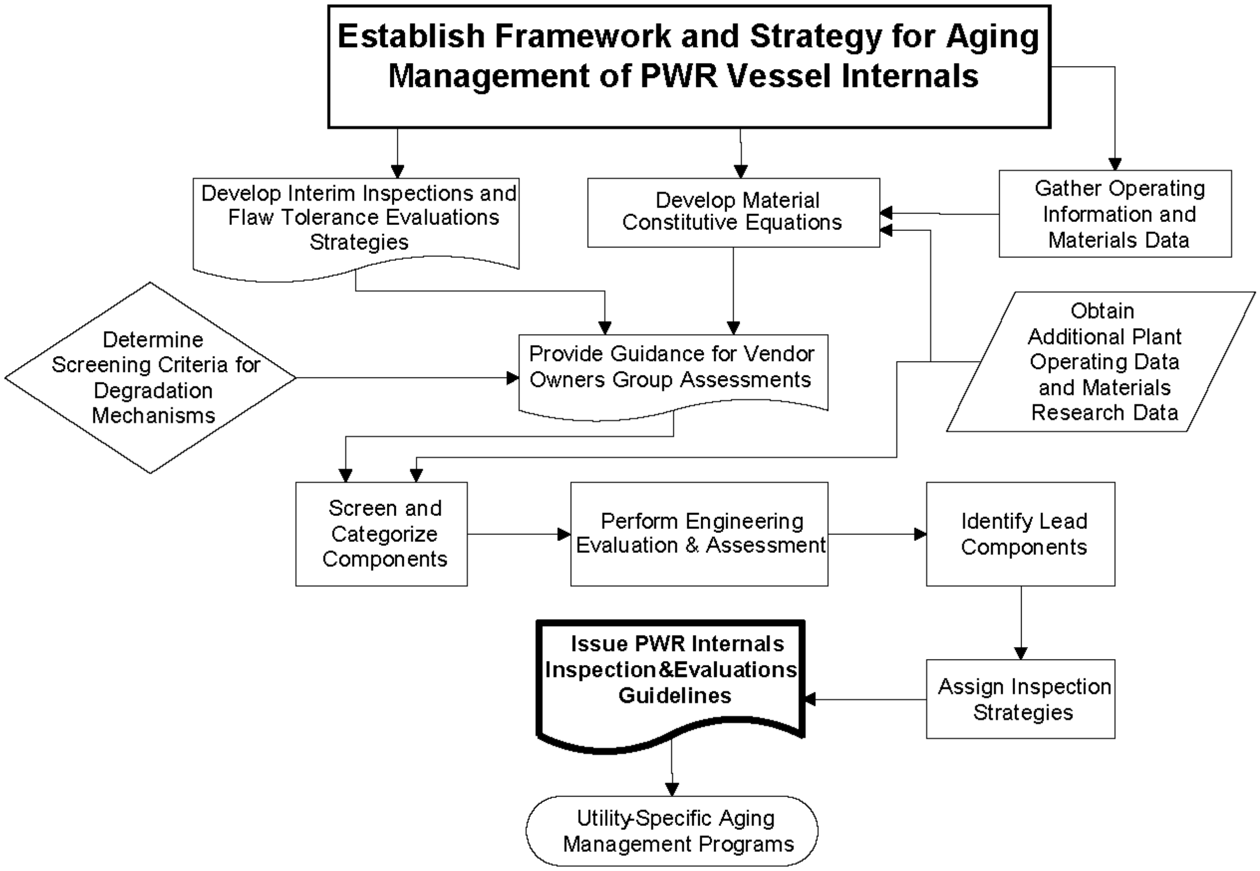

Figure 2-1
MRP Framework and Strategy for Aging Management of PWR Internals

2.2 Aging Management Strategy Development

The development of this aging management strategy combined the results of engineering evaluations and assessments with component accessibility, operating experience, existing evaluations, and prior examination results. The goal of this development has been to provide reasonable assurance for safely and economically managing aging of the PWR internals [13 and 14]. This process permitted further categorization of PWR internals into functional groups. Figure 2-2 shows the links between the categorization based on screening criteria, the engineering evaluations and safety assessments, the aging management strategy development, and the I&E guidelines. The ultimate result of the process was to assign the components into Primary, Expansion, Existing Programs, and No Additional Measures groups, with appropriate recommendations to support AMP development. Complete definitions of these four groups are provided in Section 3.3.1.

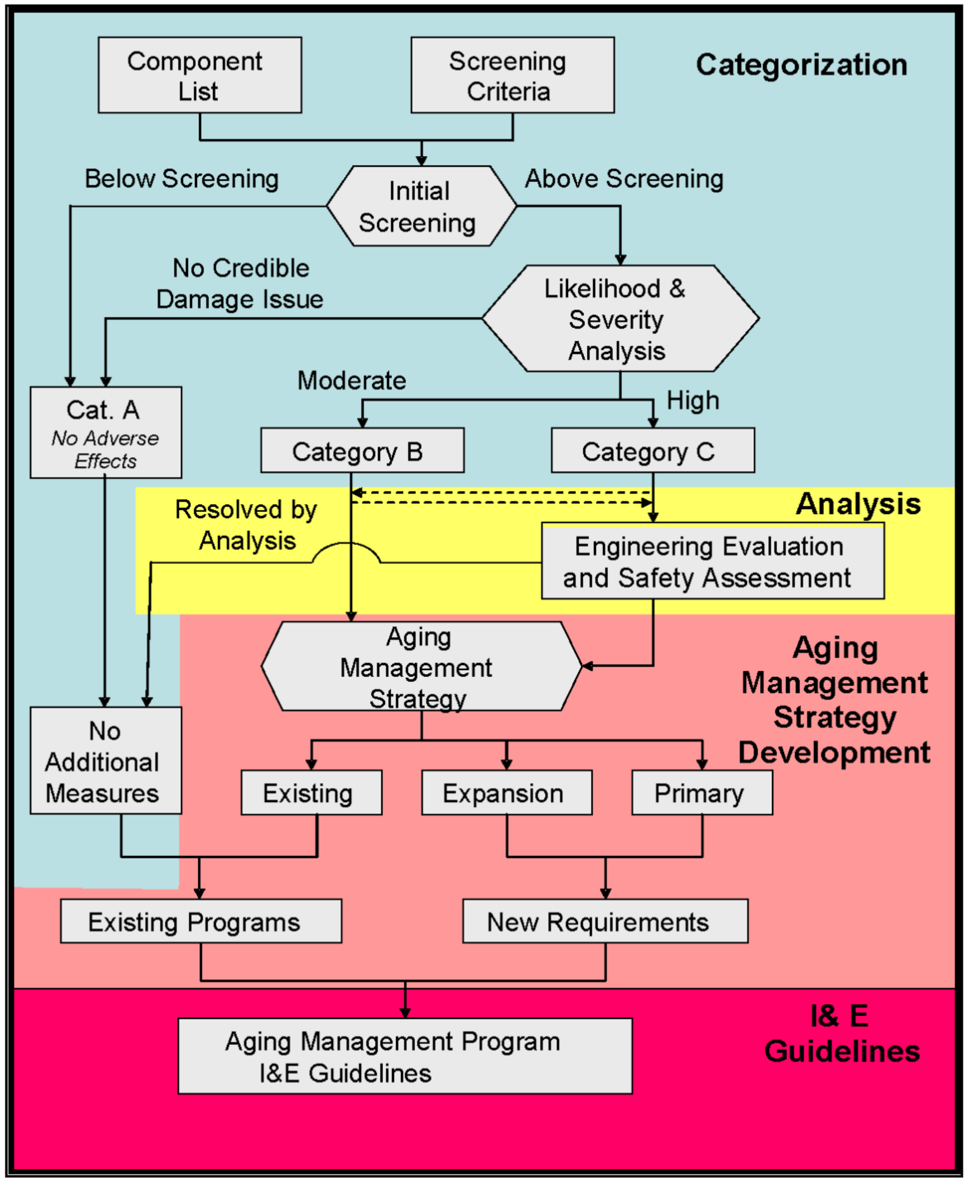

Figure 2-2
Links between Categorization, Engineering Evaluations and Assessments, Aging Management Strategy Development and the I&E Guidelines

2.3 Scope

These guidelines are intended to prescribe programs and activities that will assure the long-term safe and reliable operation of PWR internals as they age. As appropriately noted, the guidelines have requirements for both the original and the renewed licensing term.

These guidelines are applicable to the reactor internal structural components; they do not address fuel assemblies, or reactivity control assemblies. Furthermore, these guidelines are intended for operating commercial pressurized water reactors

operated as base load generation units. These guidelines do NOT supersede or modify any plant-specific commitments without specific approval to do so by the regulatory body (commitments related to 10 CFR 54, ASME Section XI, etc.). These guidelines do NOT apply to new plants beginning construction after calendar year 2007.

Section 3 provides a brief overview of currently licensed PWR internals – designed and manufactured by B&W, CE, and Westinghouse – that further defines the scope of these I&E guidelines. Section 4 identifies the components and inspection requirements. The examination acceptance criteria and the expansion criteria for the primary/expansion links are described in Section 5. Section 6 is for information and identifies various options that are available for the disposition of conditions detected during examinations (Section 4) that are unable to satisfy the examination acceptance criteria (Section 5).

The implementation of these guidelines is governed by the Materials Guidelines Implementation Protocol (Appendix B) of NEI 03-08 [1]. The Mandatory and Needed requirements are summarized in Section 7.

2.4 Guideline Applicability

The guidelines are intended to serve as the primary basis for owner preparation of an engineering program for managing aging in reactor internal components in accordance with the requirement cited in Section 7. The guidelines also serve as the primary basis for preparation of an AMP for reactor internal components to support license renewal. It is beyond the scope of the guidelines to ensure the satisfaction of every plant-specific license renewal or power uprate commitment. Plant-specific commitments remain the responsibility of the owner.

The guidelines are based on a broad set of assumptions about plant operation, which encompass the range of current plant conditions for the domestic fleet of PWRs. The engineering evaluations and assessments and the resultant supporting aging management strategies in MRP-231 [13] and MRP-232 [14] provide the basis for these guidelines. These evaluations were based on representative configurations and operational histories, which were generally conservative, but not necessarily bounding in every parameter.

Users of these guidelines are expected to confirm with reasonable assurance that each reactor managed with the guidelines satisfies the assumptions discussed as follows. General assumptions used in the analysis include:

- 30 years of operation with high-leakage core loading patterns (fresh fuel assemblies loaded in peripheral locations) followed by implementation of a low-leakage fuel management strategy for the remaining 30 years of operation, as well as the average core power levels and proximity of active fuel to the upper core support plate satisfies limits as described in Appendix B for Westinghouse/CE plants;

- The power plant has operated for the majority of its lifetime as a base-loaded unit and is currently operating as a base-loaded power plant, in that the unit operates at fixed thermal power levels and does not usually vary power on a calendar or load demand schedule;
- No design changes beyond those identified in general industry guidance or recommended by the original vendors; and
- The components and material class of each functional component are as listed in the latest revision of MRP-189 or MRP-191, as applicable to the individual plant design.

These assumptions are a conservative representation of U.S. PWR operating plants, all of which implemented low leakage core loading patterns early in operating life. The recommendations are thus applicable to all U.S. PWR operating plants as of May 2007 for the three designs identified. These guidelines are also considered applicable to plants that have replaced components or component assemblies; however, alternatives can be technically justified.

If major plant-specific differences from the inputs to the FMECA process described in MRP-189 and 191 are identified, then plant owners must determine and document the impact, if any, on the aging management strategy described herein.

Plant modifications to PWR internals (e.g., physical changes) made after calendar year 2007 should be reviewed to assess impacts on strategies contained in these guidelines. Plant modifications made or considered after this date should be reviewed to assess impacts on strategies contained in these guidelines. It is noted that some plant-specific reviews of modifications have been completed since issuance of MRP-227-A [30]; the results of these reviews have been considered in Revision 1 of these I&E guidelines.

MRP-227 originally identified that certain CE and Westinghouse PWR internals components which are subject to inspection under existing programs require further plant-specific evaluation to verify the acceptability of the existing programs, or to identify changes to the existing programs which should be implemented to manage the aging of these components for the period of extended operation. If the existing programs are not acceptable, it is necessary to identify and implement changes to the programs to manage aging of applicable components over the period of extended operation. Generically, these were components for which existing plant-specific programs other than a plant's ASME Code, Section XI program were being credited for managing aging. These components were left for plant-specific evaluation because, although the industry was able to identify that plant-specific programs already exist for the management of these components, the industry was unable to evaluate in detail the content of each facility's plant-specific program. Therefore, it is the owner's responsibility to perform a plant-specific evaluation to verify the acceptability of the existing programs, or to identify changes to the program or component replacement strategies and initiatives that should be implemented to manage the aging of these components for the period of extended operation.

In addition, users of these guidelines with CE and Westinghouse plants are encouraged to use the supplemental information provided in letters MRP-2014-017 and MRP-2014-019, which were developed to assist those plant owners in performing FMECA evaluations as well as PWROG-15032 [34] which addresses plant specific assessments of cast austenitic stainless steel (CASS) PWR internals components. These documents were previously provided to utility owners as potentially useful tools when preparing an AMP per NUREG-1801.

Section 3: Component Categorization and Aging Management Strategy Development

This section of the I&E guidelines provides a summary of the design characteristics for B&W, CE, and Westinghouse PWR internals; a summary of the screening process used for the preliminary categorization of PWR internals; and a summary of the categorization and aging management strategy development results.

3.1 Design Characteristics Summary

The functions of PWR internals are to:

1. Provide support, guidance, and protection for the reactor core;
2. Provide a passageway for the distribution of reactor coolant flow to the reactor core;
3. Provide a passageway for support, guidance, and protection for control elements and in-vessel/core instrumentation; and
4. Provide gamma and neutron shielding for the reactor vessel.

3.1.1 B&W Internals Design Characteristics

The six B&W-designed operating units share common design characteristics with minor variations. The B&W-designed PWR internals consist of two major structural assemblies that are located within, but not welded to the reactor vessel. These two major assemblies are called the plenum assembly and the core support assembly (CSA). The latter includes three principal sub-assemblies – the core support shield (CSS) assembly, the core barrel assembly, and the lower internals assembly. The general arrangement of the B&W-designed PWR internals is shown in Figure 3-1. A brief summary of the design characteristics for these internals is provided in the following sub-section. For a more complete discussion, see Reference 8.

Plenum Assembly

The plenum assembly is a cylindrical structure with perforated grid plates on top and bottom, and is comprised of: (1) the plenum cover assembly; (2) the plenum cylinder assembly; (3) the upper grid assembly; and (4) the control rod guide tube assemblies. The plenum assembly fits inside the core support shield, positions the top of the fuel assemblies, supports the control rod guide tube assemblies, and provides the core hold-down required for hydraulic lift forces. The plenum assembly also provides continuous guidance and protection for the control rods, and directs flow out of the core to reactor vessel outlet nozzles. The plenum assembly is removed at the beginning of every refueling outage to permit access to the fuel assemblies.

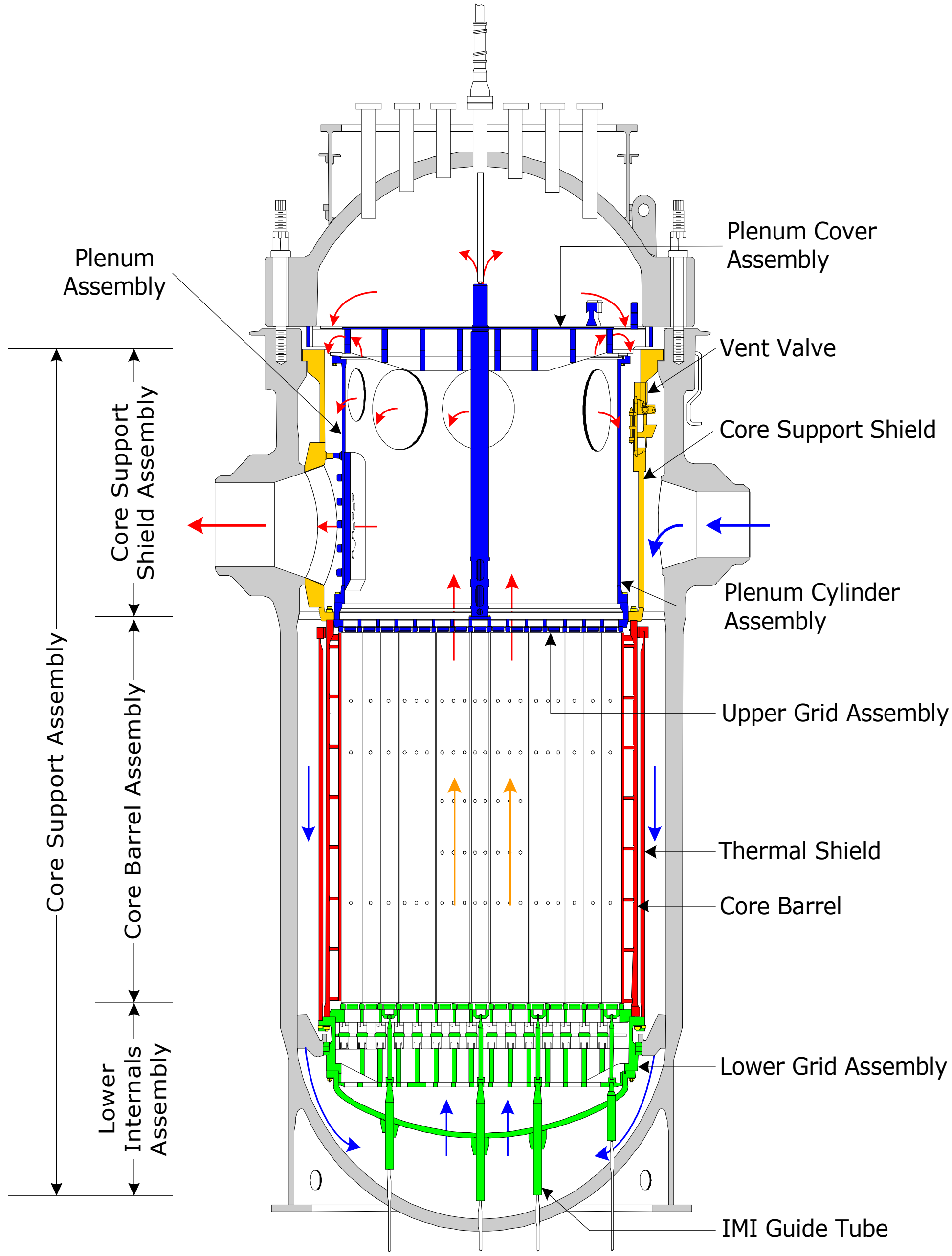

Figure 3-1
Overview of Typical B&W Internals

The plenum cover assembly is bolted to the top of the plenum cylinder, and consists of a weldment, a bottom flange, a support ring and flange, a cover plate, and lifting lugs. The plenum cover assembly provides support for the top of the control rod guide tube assemblies. The lifting lugs are used to lift the plenum assembly out of the reactor vessel.

The plenum cylinder assembly is bolted to the bottom of the plenum cover assembly and consists of a cylinder, top and bottom flanges, reinforcing plates, and round bars. Its function is to direct the flow of reactor coolant from the core region to the reactor vessel outlet nozzles.

The upper grid assembly sits inside the lower flange of the core support shield and is bolted to the plenum cylinder bottom flange. It is comprised of an upper grid ring forging, an upper grid rib section, and fuel assembly support pads. Its function is to support and provide a seating surface for the tops of the fuel assemblies located within the core barrel below, and to restrain and align the bottoms of the control rod guide tubes.

The control rod guide tube assemblies each consist of a pipe (the guide housing), a flange, spacer castings, guide tubes, and rod guide sectors. The assemblies are welded to the plenum cover plate and bolted to the upper grid assembly. Their function is to provide control rod assembly guidance, protect the control rod assembly from the effects of potential coolant cross-flow, and structurally connect the upper grid assembly to the plenum cover.

Core Support Assembly

The core support assembly is fabricated by bolting together the core support shield assembly, the core barrel assembly, and the lower internals assembly to form a tall cylinder. The core support assembly remains in place in the reactor vessel during refueling, and is removed only to perform scheduled inspections of the reactor vessel interior surfaces or of the core support assembly itself.

The top portion of the core support assembly is the core support shield assembly, a cylinder with an upper flange that rests on a circumferential support ledge in the reactor vessel closure flange, thereby supporting the entire core support assembly. It sits directly on top of the core barrel, and consists of a cylinder, top and bottom flanges, outlet nozzles, vent valve nozzles, vent valves, round bars, flow deflectors, and lifting lugs. Its function is to provide a boundary between the incoming cold reactor coolant on the outside of the cylinder and the heated reactor coolant flowing on the inside of the cylinder.

The core barrel assembly is a second flanged cylinder, with its top flange bolted to the bottom flange of the core support shield assembly and its bottom flange bolted to the top flange of the lower internals assembly. The core barrel assembly consists of a cylinder, top and bottom flanges, baffle and former plates, and a thermal shield cylinder. Its functions are to direct the flow of coolant and to support the lower internals assembly. In addition, the thermal shield reduces the amount of radiation that reaches the reactor vessel. The incoming reactor coolant is directed downward along the outside of the core barrel cylinder and upward through the fuel assemblies contained inside the core barrel. A small amount of coolant flows upward through the space between the core barrel cylinder and the baffle plates. A small portion of the coolant also runs down the annulus between the thermal shield and the core barrel cylinder, through holes drilled in the core barrel cylinder bottom flange, and then upward through the core.

The lower internals assembly consists of a lower grid assembly, a flow distributor assembly, and in-core monitoring instrumentation guide tube assemblies. The lower internals assembly is bolted to the bottom flange of the core barrel cylinder, and its function is to direct coolant flow upward through the fuel assemblies. The lower grid assembly consists of three grid structures or flow plates: (1) the lower grid rib section, (2) the flow distributor plate, and (3) the lower grid forging. Each of these flow plates has holes or flow ports to direct coolant flow upward toward the fuel assemblies.

3.1.2 CE Internals Design Characteristics

In general, the 14 operating CE-designed PWRs in the U.S. are divided into three groups: (1) those with a bolted core shroud and top-mounted in-core instrumentation (ICI); (2) those with a welded core shroud and top-mounted ICI; and (3) those with a welded core shroud and bottom-mounted ICI.

The CE-designed PWR internals consist of three major structural assemblies, plus three other sets of major components. The three major assemblies are the: (1) upper internals assembly, (2) core support barrel assembly, and (3) lower internals assembly. In addition, the three other sets of major components are the control element assembly shroud assemblies, core shroud assembly, and in-core instrumentation support system. The general arrangement of the CE-designed PWR internals is shown in Figure 3-2. A brief summary of the design characteristics for these internals is provided in the following sub-section. For a more complete discussion, see Reference 10.

Upper Internals Assembly

The upper internals assembly is located above the reactor core, within the core support barrel assembly, and is removed during refueling as a single component to provide access to the fuel assemblies. The upper internals assembly consists of the upper guide structure support plate, the fuel assembly alignment plate, the control element assembly shroud assemblies, the upper guide structure grid assembly, the upper guide structure cylinder, the in-core instrumentation support system and the hold-down ring (or expansion compensating ring). The functions of the upper internals assembly are to provide alignment and support to the fuel assemblies, to maintain control element assembly shroud spacing, to prevent movement of the fuel assemblies in the case of a severe accident condition, and to protect the control rods from cross-flow effects in the upper plenum. The flange on the upper end of the upper internals assembly rests on the core support barrel.

Core Support Barrel

The core support barrel assembly consists of the core support barrel, the core support barrel upper flange, core support barrel alignment keys, and the core support barrel snubbers. In one CE plant, a thermal shield is part of the core support barrel assembly.

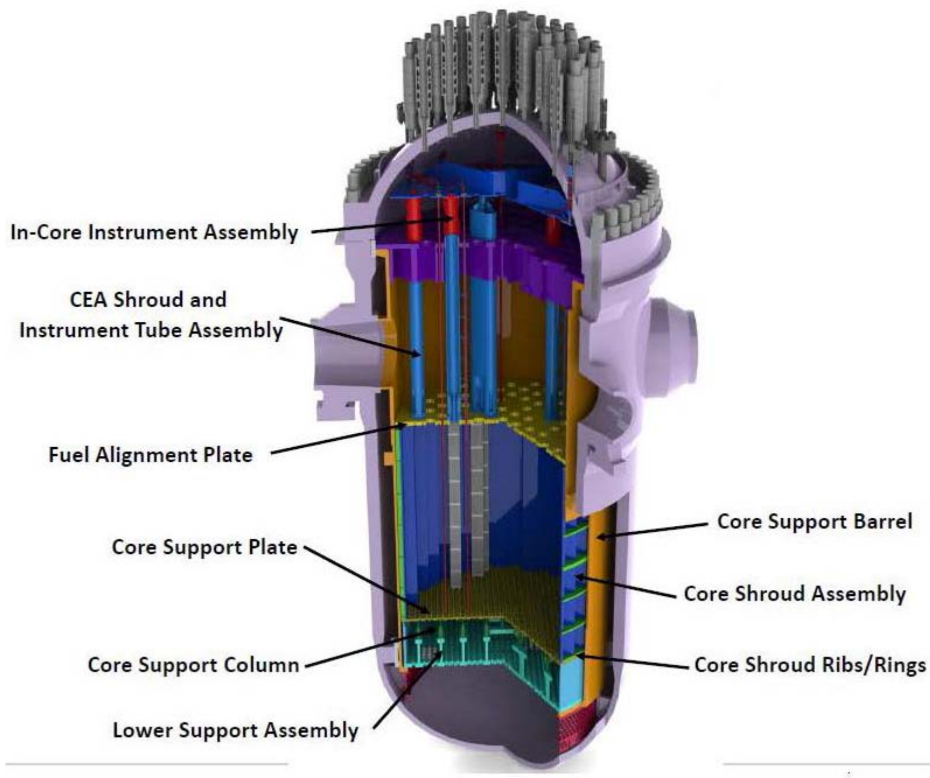

Figure 3-2
Overview of Typical CE Internals

The core support barrel is a cylinder which contains the core and other internals. Its function is to resist static loads from the fuel assemblies and other internals, and dynamic loads from normal operating hydraulic flow, seismic events, and loss-of-coolant-accident (LOCA) events. The core support barrel also supports the lower internals assembly and its core support plate, upon which the fuel assemblies rest.

The core support barrel upper flange is a thick ring that supports and suspends the core support barrel from a ledge on the reactor vessel. The core support barrel of the predominant CE design also has an inward facing lower flange for support of the lower support structure. This flange has an integral flexure ring that is welded to and provides lateral support for the lower support structure. The core support barrel is a welded construction and the susceptibility of these full-thickness welds to IASCC was researched in MRP-376 [28].

Lower Internals Assembly

The lower internals assembly consists of the core support plate, the fuel alignment pins, the core support columns, the ICI support system (some plants), and the lower support structure beam assemblies. The core support plate functions are to position and support the reactor core, and to distribute reactor coolant flow into each fuel assembly. The core support plate transmits the weight of the core to the core support barrel by means of the vertical core support

columns, an annular skirt, and the lower support structure beams. The fuel alignment pins protrude from the core support plate and provide guidance and limit lateral movement of the individual fuel assemblies. CE plants with a welded core shroud and bottom-mounted ICI have no core support plate, in which case the fuel alignment pins are attached directly to the core support deep beams.

Core Shroud Assembly

The core shroud assembly is located within the core support barrel and directly below the upper internals assembly. The core shroud assembly is attached to the core support barrel by threaded structural fasteners for those internals with a bolted core shroud and top-mounted ICI. The core shroud assembly is attached to the core support plate – an element of the lower internals assembly – by tie rods or welds for the internals with a welded core shroud and top-mounted ICI (Figure 3-3). The core shroud assembly is attached to the lower internals assembly cylinder by welding for those internals with a welded core shroud and bottom-mounted ICI (Figure 3-4). The core shroud assembly functions are to provide a boundary between reactor coolant flow on the outside of the core support barrel and the reactor coolant flow through the fuel assemblies, to limit the amount of coolant bypass flow, and to reduce the lateral motion of the fuel assemblies.

Control Element Assembly Shroud Assemblies

The control element assembly shroud assemblies consist of control element assembly shrouds, the control element assembly shroud bolts, and the control element assembly shroud extension shaft guides. The shroud tubes protect the control rods from cross-flow effects in the upper plenum. The bottom part of the shrouds is bolted at their lower end to the fuel assembly alignment plate. The extension shaft guides also protect the control rods from cross-flow effects in the upper plenum, and provide lateral support and alignment of the control element assembly extension shafts during refueling operations. The control element drive mechanisms are positioned on the reactor vessel closure head and are coupled to the control element assemblies by the control element assembly extension shafts. Control element assembly shroud assemblies are attached to the upper guide structure support plate by tie rods.

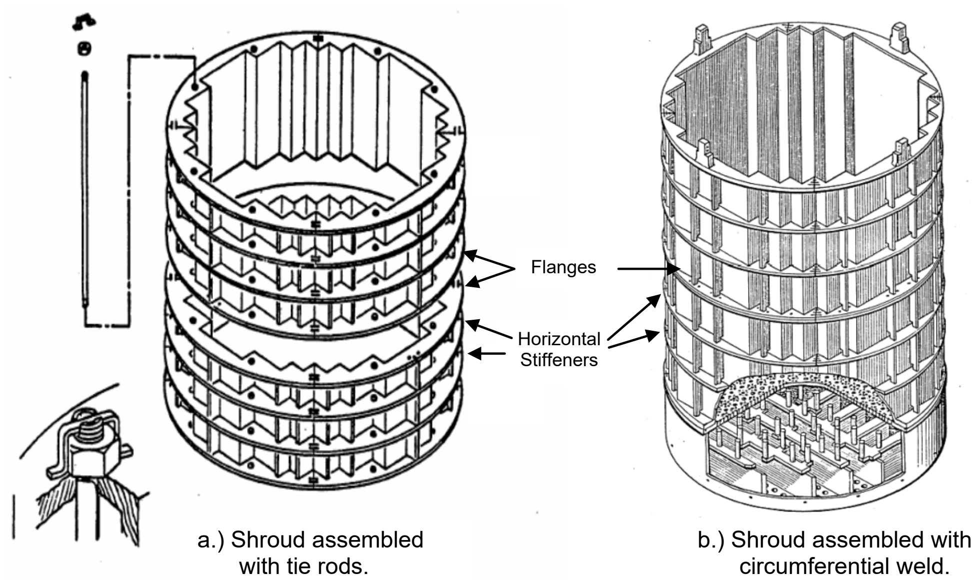

Figure 3-3
Typical CE Welded Core Shroud Designs Assembled in Two Vertical Sections (with Top- Mounted ICI)

In-Core Instrumentation Support System

The in-core instrumentation support system consists of in-core instrumentation guide tubes and components which provide support to the in-core instrumentation.

For plants with top-entry in-core instrumentation assemblies, the in-core instrumentation is inserted through the reactor vessel head through a nozzle into a guide tube. The guide tubes interface with the thimble support plate, which is perforated to fit over the control element assembly extension shaft guides, with a connection to the upper guide structure support plate. ICI thimble tube assemblies extend downward from a flanged connection at the thimble support plate (in the original design) through the fuel alignment plate and into the reactor core. The upper portion of the ICI thimble tube exists between the thimble support plate and fuel alignment plate, while the lower ICI thimble tube is the zirconium alloy portion that extends into the fuel assemblies.

For plants with bottom-entry in-core instrumentation, the guide tubes are connected to and supported by the lower internals assembly, from which the in-core instrumentation enters the core.

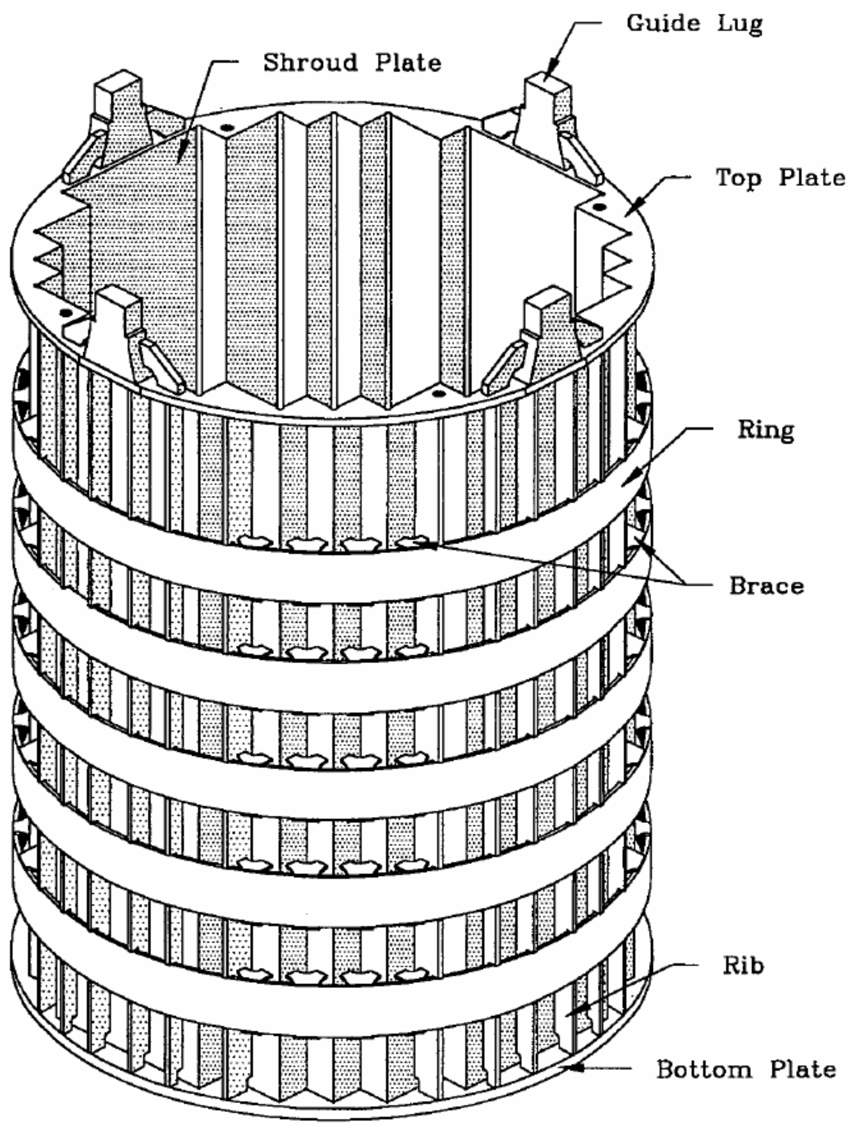

Figure 3-4
Typical CE Welded Core Shroud with Full Height Panels (with Bottom-Mounted ICI)

3.1.3 Westinghouse Internals Design Characteristics

A schematic view of a typical set of Westinghouse-designed PWR internals is shown in Figure 3-5. However, because of the significant variation in design characteristics, the operating Westinghouse PWRs in the U.S. are sub-divided into various groups, starting with the number of reactor coolant system (RCS) loops – two-loop, three-loop, and four-loop configurations. Other significant variations include the original thermal output, the baffle-barrel region flow design (downflow, upflow, and converted upflow), and upper support plate configuration. A complete set of these groups is provided in Section 4 of Reference 10.

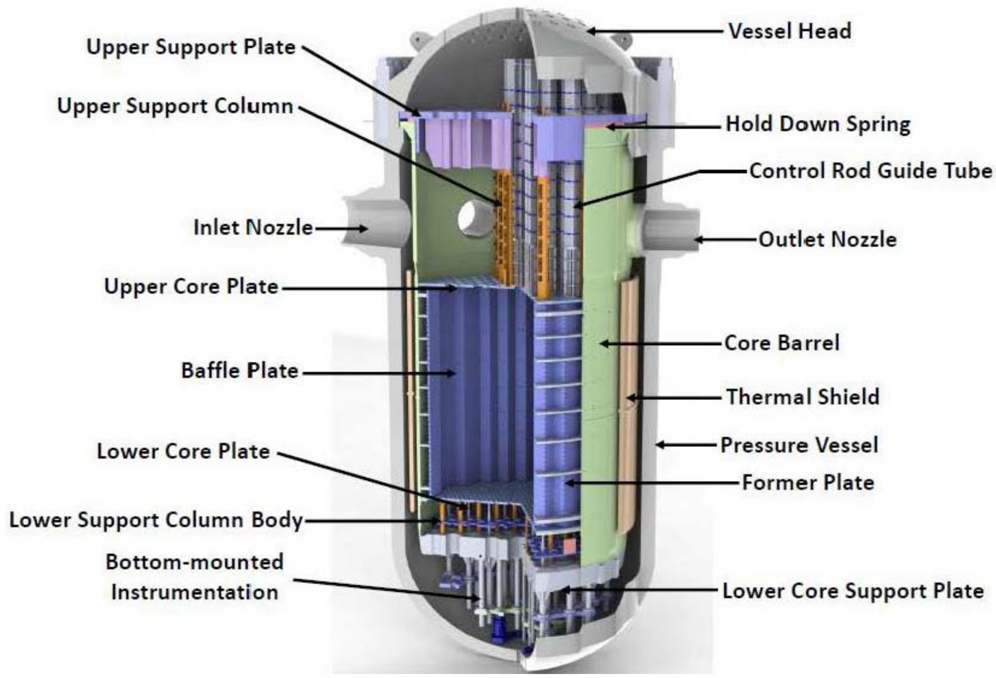

Figure 3-5
Overview of Typical Westinghouse Internals

All Westinghouse internals consist of two basic assemblies: an upper internals assembly that is removed during each refueling operation to obtain access to the reactor core and a lower internals assembly that can be removed following a complete core off-load.

Upper Internals Assembly

The major sub-assemblies that comprise the upper internals assembly are the: (1) upper core plate (UCP) and fuel alignment pins; (2) upper support column assemblies; (3) control rod guide tube assemblies and flow downcomers; (4) upper plenum; and (5) upper support plate assembly.

During reactor operation, the upper internals assembly is preloaded against the fuel assembly springs and the internals holddown springs by the reactor vessel head pressing down on the outside edge of the upper support plate (USP). The USP acts as the divider between the upper plenum and the reactor vessel head and as a relatively stiff base for the rest of the upper internals. The upper support columns and the guide tubes are attached to the USP. The UCP, in turn, is attached to the upper support columns. The USP assemblies are designated as one of three different designs: (1) a deep beam design, (2) a top hat design, or (3) an inverted top hat design.

The UCP is perforated to permit coolant to pass from the core below into the upper plenum defined by the USP and the UCP. The coolant then exits through the outlet nozzles in the core barrel. The UCP positions and laterally supports the core by fuel alignment pins extending below the plate. The UCP contacts and preloads the fuel assembly springs and thus maintains contact of the fuel assemblies with the lower core plate (LCP) during reactor operation.

The upper support columns vertically position the UCP and are designed to take the uplifting hydraulic flow loads and fuel spring loads on the UCP. The guide tubes are bolted to the USP and pinned at the UCP so they can be easily removed if replacement is desired. The guide tubes are designed to guide the control rods in and out of the fuel assemblies to control power generation. The guide tubes are also slotted in their lower sections to allow coolant exiting from the core to flow into the upper plenum.

The upper instrumentation columns are bolted to the USP. These columns support the thermocouple guide tubes that lead the thermocouples from the reactor head into the upper plenum to just above the UCP.

The UCP alignment pins locate the UCP laterally with respect to the lower internals assembly. The pins must laterally support the UCP so that the plate is free to expand radially and move axially during differential thermal expansions between the upper internals and the core barrel. The UCP alignment pins are the interfacing components between the UCP and the core barrel. The UCP alignment pins are shrunk-fit and welded into the core barrel and the core barrel bearing pad. The gap sizes between the alignment pins and the matching inserts are customized.

The USP, the upper support columns, and the UCP are typically considered core support structures.

Lower Internals Assembly

The reactor core is positioned and supported by the lower internals and upper internals assemblies. The individual fuel assemblies are positioned by fuel alignment pins in the LCP and in the UCP. These pins control the orientation of the core with respect to the lower internals and upper internals assemblies. The lower internals are aligned with the upper internals by the UCP alignment pins and secondarily by the head/vessel alignment pins. The lower internals are orientated to the vessel by the lower radial keys and by the head/vessel alignment pins. Thus, the core is aligned with the vessel by a number of interfacing components.

The lower internals assembly is supported in the vessel by clamping to a ledge below the vessel-head mating surface and closely guided at the bottom by radial support/clevis assemblies. The upper internals assembly is clamped at this same ledge by the reactor vessel head. The bottom of the upper internals assembly is closely guided by the core barrel alignment pins of the lower internals assembly.

The fuel assemblies are supported inside the lower internals assembly on top of the LCP. The LCP is elevated above the lower support forging by support columns and bolted to a ring support attached to the inside diameter of the core barrel. The support columns transmit vertical fuel assembly loads from the LCP to the much thicker lower support forging. The lower support forging is welded to and supported by the core barrel, which transmits vertical loads to the vessel through the core barrel flange.

The functions of the LCP are to position and support the core and provide a metered control of reactor coolant flow into each fuel assembly. The LCP is located near the bottom of the lower support assembly, inside the core barrel, and above the lower support forging.

The function of the lower support forging or casting is to provide support for the core. The lower support forging is attached with a full-penetration weld to the lower end of the core barrel. In this position it can provide uninterrupted support to the core. The core sits directly on the LCP, which is supported by the lower support columns that are attached to and extend above the lower support forging. Some four-loop plants employ a cast lower support instead of a forging [27]. The functions, loads, and supporting hardware are the same except for dimensions.

The primary function of the core barrel is to support the core. A large number of components are attached to either the core barrel or the core barrel flange, including the baffle/former assembly, the outlet nozzles, the neutron panel assemblies or thermal shield, the alignment pins that engage the UCP and the LCP, the lower support forging, and the LCP. The radial keys restrain large transverse motions of the core barrel but at the same time allow unrestricted radial and axial thermal expansions.

The baffle and former assembly is made up of vertical plates called baffles and horizontal support plates called formers. The baffle plates are bolted to the formers by the baffle/former bolts, and the formers are attached to the core barrel inside diameter by the barrel/former bolts. The baffle/former assembly forms the interface between the core and the core barrel. The baffles provide a barrier between the core and the former region so that a high concentration of flow in the core region can be maintained. A secondary benefit, although not a requirement of the baffles, is to reduce the neutron flux on the vessel.

Baffle plates are secured to each other at selected corners by edge bolts. In addition, in some installations, corner brackets are installed behind and bolted to the baffle plates.

The function of the core barrel outlet nozzles is to direct the reactor coolant, after it leaves the core, radially outward through the reactor vessel outlet nozzles. The core barrel outlet nozzles are located in the upper portion of the core barrel directly below the flange and are attached to the core barrel, each in line with a vessel outlet nozzle.

Additional neutron shielding of the reactor vessel is provided in the active core region by neutron panels or thermal shields that are attached to the outside of the core barrel. Specimen guides that contain specimens for determining the irradiation effects of the vessel during the life of the plant are attached to the neutron panels/thermal shields.

The flux thimble is a long, slender stainless steel tube that passes from an external seal table, through the bottom mounted nozzle penetration, through the lower internals assembly, and finally extends to the top of the fuel assembly. The flux thimble provides a path for the neutron flux detector into the core and is subjected to reactor coolant pressure and temperature on the outside surface and to atmospheric conditions on the inside. The flux thimble path from the seal table to the bottom mounted nozzles is defined by flux thimble guide tubes, which are part of the primary pressure boundary and not considered to be part of the internals. The bottom-mounted instrumentation (BMI) columns provide a path for the flux thimbles from the bottom of the vessel into the core. The BMI columns align the flux thimble path with instrumentation thimbles in the fuel assembly.

The LCP and the fuel alignment pins, the lower support forging or casting, the lower support columns, the core barrel, the core barrel flange, the radial support keys, the baffle plates, and the former plates are typically classified as core support structures.

3.2 Initial Screening Summary

This sub-section contains a summary of the initial screening of PWR internals – screening those internals on the basis of susceptibility to eight different age-related degradation mechanisms – SCC, IASCC, wear, fatigue, thermal aging embrittlement, irradiation embrittlement, void swelling, and the combination of thermal and irradiation-enhanced stress relaxation or irradiation-enhanced creep. Development and justification of the screening criteria required knowledge of the specific aging mechanisms and their effects, some engineering judgment, and the use of empirical relations where data were lacking. The full explanation of the screening criteria for the eight age-related degradation mechanisms identified for PWR internals is provided in Reference 7.

For this initial screening, the group of PWR internals that were deemed not to be susceptible to any of the eight age-related degradation mechanisms (i.e., below the screening criteria) were placed into Category A. These components are listed in previous reports for the B&W PWR designs [8] and the CE and Westinghouse PWR designs [10]. The further categorization of the components is discussed in Section 3.3.

The age-related degradation mechanisms used for the initial screening are defined in the following sub-sections. More detailed discussions of these aging mechanisms are provided in Reference 7.

3.2.1 Stress Corrosion Cracking

SCC refers to local, non-ductile cracking of a material due to a combination of tensile stress, environment, and metallurgical properties. The actual mechanism that causes SCC involves a complex interaction of environmental and metallurgical factors. The aging effect is cracking.

3.2.2 Irradiation-Assisted Stress Corrosion Cracking

IASCC is a unique form of SCC that occurs only in highly-irradiated components. The aging effect is cracking.

3.2.3 Wear

Wear is caused by the relative motion between adjacent surfaces, with the extent determined by the relative properties of the adjacent materials and their surface condition. The aging effect is loss of material.

3.2.4 Fatigue

Fatigue is defined as the structural deterioration that can occur as the result of repeated stress/strain cycles caused by fluctuating loads and temperatures. After repeated cyclic loading of sufficient magnitude, microstructural damage can accumulate, leading to macroscopic crack initiation at the most highly affected locations. Subsequent mechanical or thermal cyclic loading can lead to growth of the initiated crack. Corrosion fatigue is included in the degradation description.

Low-cycle fatigue is defined as cyclic loads that cause significant plastic strain in the highly stressed regions, where the number of applied cycles is increased to the point where a crack eventually initiates. When the cyclic loads are such that significant plastic deformation does not occur in the highly stressed regions, but the loads are of such increased frequency that a fatigue crack eventually initiates, the damage accumulated is said to have been caused by high-cycle fatigue. The aging effects of low-cycle fatigue and high-cycle fatigue are additive.

Fatigue crack initiation and growth resistance is governed by a number of material, structural and environmental factors, such as stress range, loading frequency, surface condition and presence of deleterious chemical species. Cracks typically initiate at local geometric stress concentrations, such as notches, surface defects, and structural discontinuities. The aging effect is cracking.

3.2.5 Thermal Aging Embrittlement

Thermal aging embrittlement is material embrittlement due to the exposure of delta ferrite within CASS and martensitic precipitation-hardenable (PH) stainless steel to high inservice temperatures. This can result in an increase in tensile strength, a decrease in ductility, and a loss of fracture toughness. Some degree of thermal aging embrittlement can also occur at normal operating temperatures for CASS and martensitic PH stainless steel internals. CASS

components have a duplex microstructure and are particularly susceptible to this mechanism [34]. While the mechanism's aging effects are loss of ductility and toughness, the eventual aging effect of concern is unstable crack extension when the local applied stress intensity in an existing crack exceeds the reduced fracture toughness.

3.2.6 Irradiation Embrittlement

Irradiation embrittlement is also referred to as neutron embrittlement. The mechanical properties of stainless steel and nickel-base alloys can be changed when exposed to high-energy neutrons. Such changes in mechanical properties include increasing yield strength, increasing ultimate strength, decreasing ductility, and a loss of fracture toughness [26] and [34]. The irradiation embrittlement aging mechanism is a function of both temperature and neutron fluence. While the mechanism's aging effects are loss of ductility and toughness, the eventual aging effect of concern is unstable crack extension when the local applied stress intensity in an existing crack exceeds the reduced fracture toughness.

3.2.7 Void Swelling and Irradiation Growth

Void swelling is defined as a gradual increase in the volume of a component caused by formation of microscopic cavities in the material. These cavities result from the nucleation and growth of clusters of irradiation produced vacancies. Helium produced by nuclear transmutations can have a significant impact on the nucleation and growth of cavities in the material. Void swelling may produce dimensional changes that exceed the dimensional tolerances of a component. Strain gradients produced by differential swelling in the system may produce significant stresses. Severe swelling (>5% by volume) has been correlated with extremely low fracture toughness values. Also included in this description is irradiation growth of anisotropic materials, which is known to cause significant dimensional changes in in-core instrumentation tubes fabricated from zirconium alloys. While the initial aging effect is dimensional change and distortion, severe void swelling may result in cracking under stress.

3.2.8 Thermal and Irradiation-Enhanced Stress Relaxation or Irradiation-Enhanced Creep

The loss of preload aging effect can be caused by the aging mechanisms of stress relaxation or creep. Thermal stress relaxation (or, primary creep) is defined as the unloading of preloaded components due to long-term exposure to elevated temperatures, such as seen in PWR internals. Stress relaxation occurs under conditions of constant strain where part of the elastic strain is replaced with plastic strain. Available data show that thermal stress relaxation appears to reach saturation in a short time (< 1000 hours) at PWR internals temperatures.

Creep (or more precisely, secondary creep) is a slow, time and temperature dependent, plastic deformation of materials that can occur when subjected to stress levels below the yield strength (elastic limit). Creep occurs at elevated temperatures where continuous deformation takes place under constant strain. Secondary creep in austenitic stainless steels is associated with temperatures higher than those relevant to PWR internals even after taking into account gamma heating. However, irradiation-enhanced creep (or more simply, irradiation creep) or ISR is an athermal process that depends on the neutron fluence and stress; and, it can also be affected by void swelling should it occur. The aging effect is a loss of mechanical closure integrity (or, preload) that can lead to unanticipated loading which, in turn, may eventually cause subsequent degradation by fatigue or wear and result in cracking.

3.3 Component Categorization and Aging Management Strategy Development Results Summary

3.3.1 Method and Definitions

This sub-section provides a summary of the results of the categorization of PWR internals after the initial screening. In this exercise, a Failure Modes, Effects, and Criticality Analysis (FMECA) was applied to the PWR internals. Based upon the FMECA results, the most affected PWR internals were placed into Category C (highest priority for assessments), while the components that are only moderately affected were placed into Category B (next highest priority for assessment). In addition, the FMECA process determined that some components not initially Category A (lowest priority for assessment) were sufficiently unaffected by consequences and placed into Category A. These priorities helped to define the range of the assessment sampling process.

In addition to this categorization using FMECA, a more refined assessment involved engineering evaluations and safety assessments of some of the components other than Category A components with the intent to determine the tolerance of components and systems of components to aging degradation effects. When these evaluations and assessments were completed, all PWR internals were placed into four functional groups that further defined the assessment sampling process, as summarized below:

- Primary: those PWR internals that are highly susceptible to the effects of at least one of the eight aging mechanisms were placed in the Primary group. The aging management requirements are intended to provide reasonable assurance of the continued functionality of Primary components and to predict future behavior of Expansion components as described in these I&E guidelines. Where little to no service degradation has been experienced to date and/or service degradation is not expected solely based on the aging mechanism, a sampling strategy for primary components is specified. The Primary group also includes components which have shown a degree of tolerance to a specific aging degradation effect, but for which no highly susceptible component exists or for which no highly susceptible component is accessible.

- Expansion: those PWR internals that are highly or moderately susceptible to the effects of at least one of the eight aging mechanisms, but for which engineering evaluations and safety assessments have shown a degree of tolerance to those effects, were placed in the Expansion group. The schedule for implementation of aging management requirements for Expansion components will depend on the findings from the examinations of the Primary components at individual plants. In this regard, the Expansion group also consists of those components for which an increased scope of the Primary component sample is specified based on degradation detected in the Primary sample (i.e., increased sampling of a Primary component).
- Existing Programs: those PWR internals that are susceptible to the effects of at least one of the eight aging mechanisms and for which generic and plant-specific existing AMP elements (e.g., ASME B&PV Code Section XI [2]) are capable of managing those effects, were placed in the Existing Programs group.
- No Additional Measures: those PWR internals for which the effects of all eight aging mechanisms are below the screening criteria were placed in the No Additional Measures group. Additional components were placed in the No Additional Measures group as a result of FMECA and the engineering evaluations and safety assessments. No further action is required by these guidelines for managing the aging of the No Additional Measures components.

These four categories define a sampling-based condition monitoring program, starting with periodic examinations and other inspections of highly-affected internals locations, with expanding periodic examinations and other inspections if the extent of detected degradation effects exceeds expected levels (i.e., consistent with NRC Branch Technical Position RSLB-1 in NUREG-1800). Such a sampling program provides reasonable assurance that the effects of age-related degradation mechanisms will be managed during the period of extended operation.

The categorization and analysis processes described herein are not intended to supersede any ASME B&PV Code Section XI [2] requirements. Any components that are classified as core support structures as defined in ASME B&PV Code Section XI IWA-9000, and covered by Table IWB-2500-1 Category B-N-3 [2], have requirements that remain in effect and may only be altered as allowed by 10 CFR 50.55a [4] or plant-specific licensing documentation.

3.3.2 Results of Categorization and Aging Management Strategy Development

The results of this categorization and sampling process are shown in Tables 3-1 through 3-3, which also include additional updates to reflect the NRC safety evaluation report of MRP-227 Revision 0 and the NRC safety evaluation report of MRP-227 Revision 1. In these tables, the right-hand column characterizes the final group: "P" corresponds to Primary components, "E" corresponds to

Expansion components, “X” to Existing Programs components and “N” refers to No Additional Measures components. “A”, “B” and “C” (defined in Section 2.1 and Figure 2.2) refers to the categories after the initial screening and FMECA. Additional footnotes are added to each table to delineate the adjustments stipulated in the NRC SER of MRP-227, Revision 0.

Note that the component nomenclature used in theses tables is consistent with the bases documents and may not be the same as that used in the tables of Sections 4 and 5. Also, the final grouping (P, E, N, X) listed here is an intermediate result and does not represent the final updated inspection strategy as defined for this revision of the guidelines.

Table 3-1
Final Disposition of Non-Category A Component Items and Welds in B&W PWR Internals

P (Primary), E (Expansion), X (Existing Programs), and N (No Additional Measures) "A" stands for Category "A" prior to the engineering evaluation

Component Item or Weld	Material	Initial Category	SCC	IASCC	Wear	Fatigue	TE	IE	VS	ISR/IC	Final Group
Plenum Cover Assembly											
Plenum Cover Weldment Rib Pads	304 SS	C	A	A	P	A	A	A	A	A	P
Plenum Cover Support Flange	304 SS	C	A	A	P	A	A	A	A	A	P
Plenum Cover Support Ring	304SS	C	A	A	P	A	A	A	A	A	P
Alloy X-750 Dowels-to-Plenum Cover Bottom Flange Weld	Alloy 82 Weld	B	N	A	A	A	A	A	A	A	N
Dowel-to-Rib Pad Locking Weld (ONS-1)	Alloy 69 Weld	B	N	A	A	A	A	A	A	A	N
Upper Grid Assembly											
Fuel Assembly Support Pad Cap Screws	304SS	B	A	A	N	N	A	A	A	N	N
Alloy X-750 Dowel-to-Upper Grid Rib Section Weld	Alloy 82 weld	B	N	A	A	A	A	A	A	A	N
Upper Grid Fuel Assembly Support Pads: Alloy X-750 Dowel Locking Weld	Alloy 82 weld	B	E	A	A	A	A	A	A	A	E
Control Rod Guide Tube (CRGT) Assembly											
CRGT Spacer Castings	CF3M	B	A	A	A	A	P	A	A	A	P
CRGT Tubes	304L SS	B	A	A	N	A	A	A	A	A	N
CRGT Sectors	304L SS	B	A	A	N	A	A	A	A	A	N
Core Support Shield Assembly											
CSS Top Flange	304 SS	C	A	A	P	A	A	A	A	A	P
Upper Core Barrel (UCB) Bolts (ONS-1, ONS-2, ONS-3, TMI-1)	Alloy A-286 or Alloy X-750	C	P	A	A	P	A	A	A	A	P

Table 3-1 (continued)
Final Disposition of Non-Category A Component Items and Welds in B&W PWR Internals

Component Item or Weld	Material	Initial Category	SCC	IASCC	Wear	Fatigue	TE	IE	VS	ISR/IC	Final Group
Core Support Shield Assembly											
Upper Core Barrel (UCB) Bolts (ANO-1, DB)	Alloy A-286 or Alloy X-750	C	P	A	A	A	A	A	A	A	P
Vent Valve Assembly (Notes 7 and 8)											
Vent Valve Body	CF8	B	A	A	A	A	E	A	A	A	E
CSS Vent Valve Top Retaining Ring	15-5PH	B	A	A	A	A	P	A	A	A	P
CSS Vent Valve Bottom Retaining Ring	15-5PH	B	A	A	A	A	P	A	A	A	P
Original Locking Device: Pressure Plate	304 SS	B	A	A	P	A	A	A	A	A	P
Original Locking Device: Spring Retainer	304 SS	B	A	A	P	A	A	A	A	A	P
Original Locking Device: Spring	Alloy A-286	B	A	A	P	A	A	A	A	A	P
Original Locking Device: U-Cover	304 SS	B	A	A	P	A	A	A	A	A	P
Original Locking Device: Key Ring	Type 431	B	A	A	A	A	P	A	A	A	P
Original Locking Device: Pin	Type 431	B	A	A	A	A	P	A	A	A	P
Modified Locking Device: Bolted Block	Alloy 600	B	P	A	A	A	A	A	A	A	P
Modified Locking Device: Jackscrew Crimped Locking Cup	Alloy 600	B	P	A	A	A	A	A	A	A	P
Modified Locking Device: Bolt Crimped Locking Cup	Alloy 600	B	P	A	A	A	A	A	A	A	P
Core Barrel Assembly											
Core Barrel Cylinder (including vertical and center circumferential seam welds)	304 SS, 308L SS welds	C	N or A Note 2	N	A	A	A	E	A	A	E
Core Barrel Cylinder Top Flange	304 SS	C	N	A	A	A	A	A	A	A	N
Core Barrel Cylinder Bottom Flange	304 SS	C	N	A	A	A	A	A	A	A	N
Alloy X-750 Core Barrel-to-Former Plate Dowel	Alloy X-750	B	A	A	A	A	A	N	A	A	N

Table 3-1 (continued)
Final Disposition of Non-Category A Component Items and Welds in B&W PWR Internals

Component Item or Weld	Material	Initial Category	SCC	IASCC	Wear	Fatigue	TE	IE	VS	ISR/IC	Final Group
X-750 Dowel-to-Top and Bottom Core Barrel Cylinders Fillet Weld	Alloy 82 weld	B	N	A	A	A	A	N	A	A	N
Lower Core Barrel (LCB) Bolts	Alloy A-286 or Alloy X-750	C	P	A	P	P	A	A	A	P	P
Upper Thermal Shield (UTS) Bolts	Alloy A-286 or Alloy X-750	B	E	A	A	A	A	A	A	A	E
Alloy X-750 Dowel-to-Core Barrel Cylinder Weld (Both Ends)	Alloy 82 weld	B	N	A	A	A	A	N	A	A	N
Baffle Plates	304 SS	C	A	N	A	A	A	P	N	A	P
Former Plates	304 SS	C	A	N	A	A	A	E	N	A	E
Core Barrel-to-Former (CB) Bolts	304 SS	C	A	E	E	E	A	E	A	E	E
Baffle-to-Former (FB) Bolts (Note 1)	304 SS	C	A	P	P	P	A	P	N	P	P
Internal Baffle-to-Baffle (BB) Bolts (Note 1)	304 SS	C	A	E	E	E	A	E	N	E	E
External Baffle-to-Baffle (BB) Bolts	304 SS	C	A	E	E	E	A	E	N	E	E
Accessible Locking Device and Locking Weld (FB bolts and Internal BB Bolts)	304 SS locking device, 308L SS locking weld	B	A	P or A Note 3	A	A	A	P	A	A	P
Inaccessible Locking Device and Locking Weld (CB bolts and External BB Bolts)	304 SS locking device, 308L SS locking weld	B	A	A	A	A	A	E	A	A	E
Lower Grid Assembly											
Lower Grid Rib Section	304SS	C	A	A	A	A	A	E	A	A	E

Table 3-1 (continued)
Final Disposition of Non-Category A Component Items and Welds in B&W PWR Internals

Component Item or Weld	Material	Initial Category	SCC	IASCC	Wear	Fatigue	TE	IE	VS	ISR/IC	Final Group
Lower Grid Fuel Assembly Support Pads: Pad, Pad-to-Rib Section Weld, Alloy X-750 Dowel, Cap Screw, Their Locking Welds	304SS with 308L SS weld, except Alloy X-750 dowel with Alloy 69 weld	B	A or E Note 4	A	N or A Note 5	N or A Note 5	A	E	A	N or A Note 5	E
Lower Grid Assembly Alloy X-750 Dowel-to-Guide Block Welds (Except DB-1)	Alloy 82 weld	B	P	A	A	A	A	A	A	A	P Note 9
Lower Grid Shell Forging	304 SS	C	N	A	A	A	A	A	A	A	N
Lower Grid Forging (Except ONS-1)	304 SS	B	N	A	A	A	A	A	A	A	N
Lower Grid Weldment Rib Pads (ONS-1 only)	304 SS	B	N	A	A	A	A	A	A	A	N
Alloy X-750 Bolts for Lower Grid Shock Pads (TMI-1 only)	Alloy X-750	C	P	A	A	P	A	A	A	A	P
Alloy X-750 Dowel-to-Lower Grid Shell Forging Welds	Alloy 82 weld	B	N	A	A	A	A	A	A	A	N
Alloy X-750 Dowel-to-Lower Grid Rib Section Welds	Alloy 69 weld	B	N	A	A	A	A	N	A	A	N
Lower Grid Rib-to-Shell Forging Cap Screws	304 SS	B	A	A	N	N	A	A	A	N	N
Lower Grid Support Post Pipe Cap Screws	304 SS	B	A	A	N	N	A	A	A	N	N
Lower Thermal Shield (LTS) Bolts	Alloy A-286 or Alloy X-750	B	E	A	A	A	A	A	A	A	E
Flow Distributor Assembly											
Flow Distributor (FD) Bolts	Alloy A-286 or Alloy X-750	C	P	A	A	P	A	A	A	A	P
Alloy X-750 Dowel-to-Flow Distributor Flange Welds	Alloy 82 weld	B	N	A	A	A	A	A	A	A	N

Table 3-1 (continued)
Final Disposition of Non-Category A Component Items and Welds in B&W PWR Internals

Component Item or Weld	Material	Initial Category	SCC	IASCC	Wear	Fatigue	TE	IE	VS	ISR/IC	Final Group
IMI Guide Tube Assembly											
IMI Guide Tubes	304SS	B	A	A	A	A	A	N	A	A	N
IMI Guide Tube Spiders and Spider-to-Lower Grid Rib Section Welds	CF8, 308L SS weld	B	A	A	A	A	P or A Note 6	P	A	A	P

Notes to Table 3-1:

1. Bolt overload after hard contact with the baffle and former plates is identified in Section 2.1.7. This mechanism is only applicable to the FB bolts and internal BB bolts; Primary for the FB bolts, and Expansion for the internal BB bolts.
2. Only the core barrel cylinder is categorized as No Additional Measures for SCC. The welds (vertical and center circumferential seams) are categorized as Category A for SCC.
3. Only the locking devices for the FB bolts and internal BB Bolts are categorized as Primary for IASCC. The locking device welds for the FB and internal BB bolts are Category A for IASCC.
4. Only the Alloy X-750 dowel locking weld in the listed component items for the lower grid fuel assembly support pads is susceptible to SCC and categorized as Expansion for SCC. Other component items are Category A for SCC.
5. Only the cap screws for the fuel assembly support pads are categorized as No Additional Measures for ISR/IC/fatigue/wear. The other component items and welds are Category A for ISR/IC/fatigue/wear.
6. Only the spiders are categorized as Primary for TE. The welds are Category A for TE.
7. As of May 2014, TMI-1 and DB have been verified to have original vent valve locking devices only, while ONS-1, ONS-2, ONS-3, and ANO-1 have both original and modified vent valve locking devices installed.
8. A detailed review of the ANO-1 fabrication records and/or field change packages to identify all component items and welds associated with the vent valve locking devices has not been performed. Therefore, the vent valve locking device component items and welds listed here are based on the results of the ONS-1, ONS-2, ONS-3, TMI-1, and DB evaluations.
9. For the alternate configuration at DB, the difference in configuration shall be addressed on a unit-specific basis.

Table 3-2
Final Disposition of CE Internals

Component	Material	Initial Category	SCC	IASCC	Wear	Fatigue	TE (Note 1)	IE (Note 1)	VS	ISR and IC	Final Group
Upper Internals Assembly											
Fuel Alignment Plate (Core Shrouds with Full-Height Shroud Plates)	304 SS	B	N	A	N	P	A	A	A	A	P
Lower Support Structure											
Core Support Plate	304 SS 304L SS	C	N	N	N	P	A	P	A	A	P
Fuel Alignment Pins (Core Shrouds with Full-Height Shroud Plates)	A286 SS	C	A	X	X	X	A	X	A	X	X
Core Support Columns	304 SS	B	E Note 5	E Note 5	A	E Note 5	A	E Note 5	A	A	E Note 5
Core Support Columns	CF8	B	E Note 5	E Note 5	A	E Note 5	E Note 5	E Note 5	A	A	E Note 5
Core Support Deep Beams (Core Shrouds with Full-Height Shroud Plates)	304 SS	C	X	X	A	P	A	P	A	A	P
Core Support Column Bolts	316 SS	B	A	E	N	E	A	E	A	N	E
Lower Core Support Beams	304 SS	A	E Note 6	A	A	E Note 6	A	A	A	A	E Note 6
Control Element Assembly (CEA)-Shroud Assemblies											
Instrument Tubes	304 SS	B	P	A	A	P	A	A	A	A	P
Core Support Barrel Assembly											
Upper Cylinder (including girth welds)	304 SS	B	E	A	A	A	A	A	A	A	E

Table 3-2 (continued)
Final Disposition of CE Internals

Component	Material	Initial Category	SCC	IASCC	Wear	Fatigue	TE (Note 1)	IE (Note 1)	VS	ISR and IC	Final Group
Lower Cylinder Girth Welds	304 SS	C	P Note 4	P Note 7	A	A	A	P Note 4	A	A	P Note 4
Lower Cylinder Axial Welds	304 SS	C	E	E	A	A	A	E	A	A	E
Upper Core Barrel Flange Weld	304 SS	B	P	A	X	A	A	A	A	A	P
Upper Core Barrel Flange	304 SS	A	N	N	A	N	N	N	N	N	N
Lower Core Barrel Flange	304 SS	B	E	A	A	E	A	A	A	A	E
Lower Core Barrel Flange Weld	304 SS	B	E	A	A	P	A	A	A	A	P
Thermal Shield Positioning Pins (Note 2)	UNS S21800	B	A	A	N	N	A	A	A	N	N
Core Shroud Assembly											
Shroud Plates (Bolted) (Entire Assembly)	304 SS	C	N	E	A	A	A	P	P	A	P
Shroud Plates (Welded)	304 SS	C	N	P	A	A	A	P	P	A	P
Former Plates (Bolted) (Entire Assembly)	304 SS	B	N	E	A	A	A	P	P	A	P
Former Plates (Welded)	304 SS	B	N	P	A	A	A	P	P	A	P
Ribs	304 SS	B	N	E	A	A	A	E	N	A	E
Rings (Core Shrouds with Full-Height Shroud Plates)	304 SS	B	N	E	A	A	A	E	N	A	E
Core Shroud Bolts	316 SS	B	A	P	N	N	A	P	P	P	P
Barrel-Core Shroud Bolts	316 SS	B	A	E	N	N	A	E	A	E	E
Core Shroud Tie Rods	348 SS	B	A	A	N	N	A	N	A	N	N
Core Shroud Tie Rod Nuts	316 SS	B	A	A	N	N	A	N	A	N	N
Guide Lug Insert Bolts (Note 3)	A286 SS	B	A	A	X	X	A	A	A	X	X

Table 3-2 (continued)
Final Disposition of CE Internals

Component	Material	Initial Category	SCC	IASCC	Wear	Fatigue	TE (Note 1)	IE (Note 1)	VS	ISR and IC	Final Group
In-Core Instrumentation (ICI)											
ICI Thimble Tubes-Lower	Zircaloy-4	C	A	A	X	A	A	A	A	A	X

Notes to Table 3-2:

1. The significance of thermal and irradiation embrittlement is directly related to the probability of a flaw existing in the component. There are no recommendations for inspection to determine embrittlement level because these mechanisms cannot be directly observed. However, potential embrittlement must be considered in flaw tolerance evaluations.
2. One plant has an existing program for this item.
3. Bolt deterioration may lead to degradation in lug fixtures. Inspection recommendations relate to the entire guide lug fixture.
4. Upgraded to "Primary" from "Expansion" in accordance with the NRC SER on MRP-227, Revision 0 [29].
5. Upgraded to "Primary" from "Expansion" in accordance with the NRC SER on MRP-227, Revision 0 [29], but per the NRC SER on MRP-227, Revision 1 [35], the core support columns have been assigned to the Expansion category.
6. Upgraded to "Expansion" from "No Additional Measures" in accordance with the NRC SER on MRP-227, Revision 0 [29].
7. Mechanism "IASCC" upgraded from "No Additional Measures" to "Primary" for lower cylinder welds in accordance with the NRC SER on MRP-227, Revision 0 [29].

Table 3-3
Final Disposition of Westinghouse Internals

Component	Material	Initial Category	SCC	IASCC	Wear	Fatigue	TE (Note 1)	IE (Note 1)	VS	ISR and IC	Final Group
Control Rod Guide Tube Assembly											
Lower Flanges	CF8	B	P	A	A	P	P	P	A	A	P
Guide Plates (Cards)	304 SS	C	N	A	P	N	A	A	A	A	P
C-Tubes (Note 2)	304 SS	C	A	A	P	A	A	A	A	A	N
Sheaths (Note 2)	304 SS	C	A	A	P	A	A	A	A	A	N
Guide Tube Support Pins	Alloy X-750	C	X	A	X	X	A	A	A	N	X
Upper Internals Assembly											
Upper Support Ring or Skirt	304 SS	B	E	A	A	X	A	A	A	A	X
Upper Core Plate	304 SS	A	A	A	E Note 5	E Note 5	A	A	A	A	E Note 5
Baffle-Former Assembly											
Baffle-Edge Bolts	316 SS, 347 SS	C	A	P	N	P	A	P	P	P	P
Baffle Plates and Former Plates (Note 3)	304 SS	B	A	N	A	A	A	N	P	A	P
Baffle-Former Bolts	316 SS, 347 SS	C	A	P	N	P	A	P	P	P	P
Barrel-Former Bolts	316 SS, 347 SS	C	A	E	N	E	A	E	E	E	E
Bottom Mounted Instrumentation System											
BMI Column Bodies	304 SS	B	N	N	A	E	A	E	N	A	E
BMI Column Collars	304 SS	B	A	N	A	A	A	N	N	A	N
BMI Column Cruciforms	CF8	B	A	N	A	A	N	N	N	A	N
BMI Column Extension Tubes	304 SS	B	N	N	A	A	A	N	N	A	N

Table 3-3 (continued)
Final Disposition of Westinghouse Internals

Component	Material	Initial Category	SCC	IASCC	Wear	Fatigue	TE (Note 1)	IE (Note 1)	VS	ISR and IC	Final Group
Flux Thimble Tube Plugs	304 SS	B	N	N	A	A	A	N	N	A	N
Flux Thimbles (Tubes)	316 SS	C	N	N	X	A	A	N	N	A	X
Core Barrel Assembly											
Core Barrel Flange	304 SS	B	E	A	X	A	A	A	A	A	X
Core Barrel Outlet Nozzle Welds	304 SS	B	E	A	A	E	A	A	A	A	E
Core Barrel Girth Welds (Upper / Mid-Core, highest fluence axial location)	304 SS	C	P Note 6	P Note 6	A	A	A	P Note 6	A	A	P Note 6
Core Barrel Girth Welds (Lower Flange)	304 SS	C	P	P	A	A	A	X	A	A	X
Core Barrel Axial Welds	304 SS	C	E	E	A	A	A	E	A	A	E
Upper Core Barrel Flange Weld	304 SS	C	P	E	A	A	A	A	A	A	P
Lower Internals Assembly											
Lower Core Plate	304 SS	C	N	X	X	X	A	X	N	A	X
XL Lower Core Plate	304 SS	C	N	X	X	X	A	X	A	A	X
Lower Support Casting	CF8	A	A	A	A	A	E Note 5	A	A	A	E Note 5
Lower Support Forging	304 SS	A	A	A	A	A	A	A	A	A	E Note 5
Lower Support Assembly											
Lower Support Column Bodies	CF8	B	A	E	A	A	N	E	N	A	E
Lower Support Column Bodies	304 SS	B	A	E	A	A	A	E	N	A	E
Lower Support Column Bolts	304 SS	B	A	E	N	E	A	E	N	E	E
Thermal Shield Assembly											
Thermal Shield Flexures	304 SS	B	A	N	P	P	A	N	A	N	P

Table 3-3 (continued)
Final Disposition of Westinghouse Internals

Component	Material	Initial Category	SCC	IASCC	Wear	Fatigue	TE (Note 1)	IE (Note 1)	VS	ISR and IC	Final Group
Alignment and Interfacing Components											
Internals Hold Down Spring (Note 4)	304 SS	B	A	A	P	A	A	A	A	A	P
Upper Core Plate Alignment Pins	304 SS	B	X	A	X	A	A	A	A	A	X
Clevis Insert Bolts	Alloy X-750	B	A	A	X	A	A	A	A	A	X

Notes to Table 3-3:

1. The significance of thermal and irradiation embrittlement is directly related to the probability of a flaw existing in the component. There are no recommendations for inspection to determine embrittlement level because these mechanisms cannot be directly observed. However, potential embrittlement must be considered in flaw tolerance evaluations.
2. Some of the items in the control rod guide tube (CRGT) assembly, namely the C-tubes and sheaths, have been placed in the No Additional Measures group, because decisions on remediation of wear and degradation in the CRGT assembly will be based only on the conditions detected in the Primary CRGT item, the guide tubes (cards).
3. The concern is a result of the collective interaction of all components that comprise the assembly and not strictly focused on the plates.
4. The hold-down spring does not directly degrade by wear. It first degrades by loss in preload, which leads to wear when an inadequate preload remains.
5. Upgraded to "Expansion" from "No Additional Measures" in accordance with the NRC SER [29].
6. Upgraded to "Primary" from "Expansion" in accordance with the NRC SER [29].

Section 4: PWR Internals Management Requirements

The ultimate goal of the PWR internals I&E guidelines are to monitor the condition of the internals to maintain appropriate levels of plant safety and reliability. Properly managed, the plants will fulfill their applicable license renewal commitments through application of these guidelines.

Inspection and evaluation in support of aging management requirements typically consists of the following:

- Consideration of aging mechanism susceptibility and selection of items requiring aging management;
- Selection of the type of examination or other methodologies appropriate for each applicable degradation mechanism;
- Specification of the required level of examination qualification;
- Schedule of first and frequency of any subsequent examinations;
- Item/component inspection sampling and coverage;
- Expansion of scope if sufficient evidence of degradation is observed;
- Examination acceptance criteria;
- Methods for evaluating examination results not meeting the examination acceptance criteria;
- Updating the program based on industry-wide results; and
- Contingency measures to repair, replace, or mitigate.

The listed elements of inspection and evaluation interrelate. For example, the particulars of the examination acceptance criteria may affect the rules for sampling or frequency of examination.

This section of the I&E guidelines specifies aging management requirements that are appropriate to detect the expected effects of the degradation mechanisms, are considered acceptable for the development of an engineering program, and provide for partial satisfaction of an AMP for license renewal. The goal of the engineering program is to satisfactorily manage the utility asset

and to accomplish the license renewal AMP goal, which is to ensure the continued achievement of safety-related functions and functionality of the PWR internals. The technical bases used to develop these aging management requirements are documented elsewhere [13, 14].

Some of the aging management requirements listed (e.g., examination acceptance criteria) deserve greater elaboration and are therefore discussed in Section 5.

Section 4.1 describes the overall aging management approach. Then, Section 4.2 describes the various examination methodologies, ranging from general condition visual examinations to more rigorous visual, surface, and volumetric examinations, with a final sub-section that describes physical measurement. Section 4.3 summarizes the examination requirements that are recommended for two groups of PWR internals, Primary and Expansion.

It is noted that the intent of these guidelines is to take advantage of inspection/examination scheduling flexibility coinciding with removal of the core support structure (particularly the core barrel) in conjunction with existing protocols and flexibility associated with inspection intervals for ASME required periodic inspections (e.g., 10-year ISI exams). In all cases, the flexibility of the scheduling process and protocol for the timing of ASME inspection intervals as delineated in the Section 4 tables is intended to be the driving factor; thus the PWR internals inspections and examinations specified within this guideline benefit from the scheduling flexibility allowed by ASME Code Section XI IWA-2430. For example, the term "10-year ISI interval" is intended to mean the plant's existing schedule associated with removal of the core barrel.

If components are repaired, modified or replaced such that the effects of aging are mitigated, then the demonstration of the adequacy of repair, replacement, or modification activities is the responsibility of the owner. In addition, repair, replacement or modification activities may warrant revision to the scope and/or frequency of the generic requirements stated in these guidelines. This includes re-establishing the technical basis for the replaced components (if not fully mitigated) and the technical basis for examination (and reinspection interval) of any linked Expansion components, which was developed on the basis of expert panel solicitation. Individual utilities will be responsible for the technical justification of such activities to demonstrate their acceptability for different requirements than those stated in these guidelines.

The requirements for the PWR internals components in the Existing Programs group are described in Section 4.4. As described in Section 4.5, those PWR internals components in the No Additional Measures group require no further actions with respect to management of aging degradation, other than to continue any existing requirements that affect those components.

4.1 Aging Management Approach

The aging management approach for PWR internals consists of four major elements:

1. Component categorization and aging management strategy development, including a description of the overall sampling process;
2. Identification of aging management methods for PWR internals that are both appropriate and based on an adequate level of applicable experience;
3. Qualification of the recommended methods that is based on adequate technical justification; and
4. Implementation of the recommendations based on the NEI 03-08 Guideline for the Management of Materials Issues [1].

Each element in the approach is described in the following paragraphs.

4.1.1 PWR Internals Categorization and Aging Management Strategy Development

The PWR internals categorization and aging management strategy development were summarized in Section 3, including the overall sampling strategy.

4.1.2 Selection of Established Aging Management Methods

The second part of the aging management approach involved the selection of aging management methods for the PWR internals. The criteria for selection were based on:

- The methods should be appropriate for the characterization of particular age-related degradation effects; and
- The methods should concentrate on techniques that have been subject to widespread application.

For these two reasons, the selected aging management methods emphasize existing, well-proven techniques that have been subject to widespread, relevant application. These methods are described in Section 4.2.

4.1.3 Aging Management Method Qualification

An extensive experience base for the aging management methods described in this section of the I&E guidelines permits selection of known aging management methods. Many inspections specified herein are remote visual examinations, whether visual VT-1, EVT-1 or VT-3. Remote visual examinations must meet the additional generic requirements of the MRP-228 Inspection Standard [3] for equipment and training of personnel, and in the case of visual EVT-1, a surface condition assessment and limitations on camera angle and scan speed. All other methodologies specified herein already have well established procedural qualifications, such as volumetric examination of bolting [16]. Thus the level

of procedural qualification for examinations other than remote visual is limited to technical justification. This level of qualification is appropriate. Failures of internals do not result in pressure boundary failures. Internals are either of robust design resulting in flaw tolerance well above the detection level that can be established via technical justification, or consist of assemblies for which a single (or often multiple) component item failure does not prevent the assembly from performing its function.

The MRP-228 Inspection Standard [3] provides detailed guidance for conducting and justifying the selected examination techniques and the technical justifications required for different examination methods and component configurations.

4.1.4 Implementation of Aging Management Requirements

Information on the implementation of the aging management requirements is provided in Section 7 of these I&E guidelines.

4.2 Inspection Methods for Monitoring Aging

The inspection methods described in these guidelines include visual examinations, surface examinations, volumetric examinations, and physical measurements. Each of these methods is suitable for managing the effects of one or more aging degradation mechanisms for PWR internals, depending upon:

- Tolerance of the component functionality to the progression of particular effects;
- Accessibility of the component by the equipment needed for the examination; and
- Suitability of the equipment for detecting the particular effect.

Where appropriate, the examination methods selected for use in these guidelines are as specified in U.S. Nuclear Regulatory Commission (NRC) approved editions and addenda of ASME B&PV Code Section XI [2], as specified and conditioned by 10 CFR 50.55a.

4.2.1 Visual (VT-3) Examination

One examination method selected for use in these guidelines, which has an extensive history of use for PWR internals, is visual (VT-3) examination. Such visual examinations are exclusively relied upon for detection of general degradation of PWR internals subject to Table IWB-2500-1 Category B-N-3 [2] requirements. Visual (VT-3) examinations are conducted to determine the general mechanical and structural condition of components by detecting discontinuities and imperfections, such as loss of integrity at bolted or welded connections, loose or missing parts, debris, corrosion, wear, or erosion; and by

identifying conditions that could affect operational or functional adequacy of components. This type of examination has been determined to be acceptable for the continued monitoring of many of the internals within the scope of these guidelines.

When specified in these guidelines, a visual (VT-3) examination is conducted in accordance with the requirements of the MRP-228 Inspection Standard [3]. Visual (VT-3) examinations of internals are conducted using remote examination techniques, because of personnel radiation exposure issues.

A large amount of industry experience is available relative to the application of visual (VT-3) examination procedures for examining PWR internals; however, implementation of character height requirements for VT-3 is relatively new. In fact, the resolution requirements for both visual VT-3 and visual VT-1 were both changed to more exacting standards with the 1998 ASME Code Section XI. The change from no character height for visual (VT-3) and a 1/32″ black line resolution for visual (VT-1) to 0.106″ and 0.044″ character discrimination has resulted in the relative equivalence of the current visual (VT-3) to the previous VT-1 requirements and the current visual (VT-1) to the previous EVT-1 (1/2-mil wire) resolution. Thus, the VT-3 required by these guidelines has greater detection capability than most of the Table IWB-2500-1 B-N-3 [2] examinations previously conducted.

The visual VT-3 examination has been shown to be capable of detecting IGSCC cracking in BWR shrouds and other internals components. In all cases where visual (VT-3) has been selected as the examination method to detect cracking in these guidelines, the intent is to find cracked, fractured or separated welds or base material such as locking tack welds, small component item assembly welds, cracked ligaments of drilled plates, or missing or fractured small component items. Numerous examples exist in the BWR fleet experience where VT-3 has demonstrated the capability to detect cracking (e.g., as discussed in utility correspondence documented in ML14139A178 [36]).

In addition to numerous BWR remote visual examination examples mentioned above, direct visual (VT-3) examinations of component supports using ASME B&PV Code Section XI Article IWF requirements over the past forty years have demonstrated the capability to detect relevant conditions similar to those being sought in these guidelines. Many of the visual (VT-3) relevant and non-relevant conditions noted in IWF-3410 are directly transferrable to the guidance employed in the MRP-228 Inspection Standard. Furthermore, in much the way that an examiner conducts a visual (VT-3) exam of a structural support component, upon detection of a potential indication, an examiner is motivated to observe the area of interest more closely; thus, the remote VT-3 examiner would simply “zoom-in” or move the camera closer to the anomaly to obtain a better view, understand, and document the condition.

4.2.2 Visual (VT-1 and EVT-1) Examinations

Other examination methods selected for use in these guidelines are visual (VT-1 and EVT-1) examinations. The visual (VT-1) examinations and the enhanced visual (EVT-1) examination were selected where a greater degree of detection capability, as well as sizing capability, is required – over and above the capability inherent in visual (VT-3) examinations to manage the aging effects. Unlike the detection of general degradation conditions by visual (VT-3) examination, visual (VT-1) and enhanced visual (EVT-1) examinations are conducted to detect and size discontinuities and imperfections on the surface of components, including such conditions as cracks, wear, corrosion, or erosion. Specifically, VT-1 is used for the detection and sizing of surface discontinuities such as gaps, while EVT-1 is used for the detection of surface breaking flaws.

When specified in these guidelines, a visual (VT-1) examination is conducted in accordance with the requirements of the MRP-228 Inspection Standard [3]. Enhanced visual (EVT-1) examination is also conducted in accordance with the requirements described for visual (VT-1) examination with additional requirements (such as camera scanning speed) as specified in the MRP-228 Inspection Standard [3]. When adjacent base metal is specified in the inspection coverage requirement, it is intended to include the base metal heat-affected zone (HAZ) adjacent to the weld. The adjacent base metal to be examined is defined in Section 2.3.6.4 of MRP-228 and is referenced in the Section 4 Primary and Expansion component tables in this document.

From a historical perspective, as with visual (VT-3) examination, the current ASME B&PV Code [2] requirements for visual (VT-1) examination became more rigorous than those defined in previous ASME B&PV Code versions. Many previous VT-1 examinations were only required to discern a 1/32" black line on a gray background. These limitations led the NRC and industry to adopt modified visual examinations for use in detecting flaws discovered in boiling water reactor (BWR) internals. The most recent research conducted by the EPRI Non-Destructive Examination (NDE) Center established the VT-1 character heights specified in Reference 2 as equally or better able to detect the degradation effects than the modified visual examination requirements developed previously [17]. The current ASME B&PV Code character discrimination has thus been adopted as the standard for the industry's enhanced visual EVT-1 with the enhancement above ASME B&PV Code requirements becoming a surface cleaning assessment and a more stringent acceptable camera angle radius intended to enhance the ability to detect tight surface breaking flaws, like those indicative of SCC.

4.2.3 Surface Examination

To further characterize discontinuities on the surface of components, surface examination can supplement either visual (VT-3) or (VT-1/EVT-1) examinations specified in these guidelines. This supplemental examination may thus be used to reject or accept relevant indications. A surface examination is an examination that indicates the presence of surface discontinuities, and the

ASME B&PV Code [2] lists magnetic particle, liquid penetrant, eddy current, and ultrasonic examination methods as surface examination alternatives. Here, only the electromagnetic testing (ET), also called eddy current surface examination method, is covered.

When selected for use as a supplemental examination to examinations performed in these guidelines, an ET examination is conducted in accordance with the requirements of the MRP-228 Inspection Standard [3].

ET examination is widely used for heat exchanger tubing inspections. Eddy currents are induced in the inspected object by electromagnetic coils, with disruptions in the eddy current flow caused by surface or near-surface anomalies detected by suitable instrumentation. Industry experience with ET examination is robust, especially in the aerospace and petroleum refinery industries. The experience base for PWR nuclear systems is also robust, in particular for examination of steam generator, flux thimble, and heat exchanger tubing.

4.2.4 Volumetric Examination

Another method selected for use in these guidelines is volumetric examination. An ultrasonic examination (UT) was selected where visual or surface examination is unable to detect the effect of the age-related degradation for some PWR internals. For example, IASCC in baffle/former bolts may occur underneath the bolt head, and will be undetectable by visual or surface examination unless the bolt is removed and subject to examination over its entire length.

When specified in these guidelines, an ultrasonic examination (UT) is conducted in accordance with the requirements of the MRP-228 Inspection Standard [3].

While UT has only been selected for use in these guidelines for detection of aging effects in bolting, UT is also permissible as an alternative or supplement to the specified visual examinations for other configurations such as plates and welds. This is consistent with Reference 2.

The industry has had extensive experience with the application of ultrasonic examination (UT) to PWR internals bolts, pins, and fasteners, in particular with baffle/former bolting examinations. The industry also has extensive experience in applying UT to BWR internals to detect intergranular stress corrosion cracking (IGSCC) in stainless steel and nickel-base welded plates, stainless steel internals piping, and nickel-base forgings and bolting.

4.2.5 Physical Measurements

The effects of loss of material caused by wear, the loss of pre-load or clamping force caused by such mechanisms as thermal and irradiation-enhanced stress relaxation, and excessive distortion or deflection caused by void swelling can be managed in some cases by physical measurements.

In some cases, these effects may involve changes in clearances, settings, and physical displacements that can be monitored by visual means, supplemented by physical measurements that characterize the magnitude of the effects. This methodology may be used in conjunction with visual examination, which includes "verifying parameters, such as clearances, settings, and physical displacements." The measurement of these parameters and their comparison to prescribed limits extends beyond visual (VT-3) examination, and will be referred to as "physical measurement of the effects of degradation."

4.2.6 Subsequent Examination Intervals

Where "ten-year intervals" or "the 10-year ISI" are stipulated in this document, for example, as the subsequent examination frequency in the Examination Method/Frequency column of Tables 4-1 through 4-6, this re-examination frequency is intended to be nominally 10 years consistent with the examination flexibility allowed by ASME Code Section XI Article IWA-2430.

4.2.7 Coordination with ASME Section XI Code Requirements

The MRP-227 PWR internals inspections do NOT supersede any ASME B&PV Code Section XI requirements. It is expected that owners may implement the inspection requirements of these I&E guidelines in coordination with inspections required under the ASME Section XI Code for the PWR internals, as specified in and conditioned by the Code of Federal Regulations 10CFR 50.55a. While the inspection requirements of these I&E guidelines have been developed with recognition of such existing Code inspection requirements, these I&E guidelines do not relieve the owner of responsibility to implement all of the Code inspection requirements. Relief from those requirements may be granted only under the provisions of 10CFR 50.55a. It is the responsibility of the owner to determine if certain inspections performed per these I&E guidelines may be credited toward the inspection coverage required under ASME Section XI Code, if so desired. Additionally, there may be items for which inspection coverage required by these I&E guidelines is less than the coverage required by ASME Section XI. In such cases it is the responsibility of the owner to perform any additional inspections to provide total coverage satisfying the Code requirements. These I&E guidelines take no position regarding Code classification of components. The expectation is that if a specified component is not included in the ASME Code 10-year ISI inspection, the item shall be included as an augmentation of the ISI program.

4.3 Primary and Expansion Component Requirements

The aging management requirements for Primary and Expansion PWR internals are covered in this section. As described in Section 3.3, Primary components are those for which the effects of at least one of the eight aging mechanisms is above the screening criteria, and for which additional aging management is needed to manage those effects. The particular additional aging management methods were selected from the methods described in Section 4.2. The implementation schedule for the Expansion components will depend on the findings from the

application of the additional aging management methodologies to the Primary components. The coverage requirements for Primary components may include a component sampling strategy. Where specified, the technical bases emphasize condition monitoring of a sample of components most affected by particular aging effects (Primary components), with increases in the sample population (Expansion components) if the condition monitoring programs detect aging effects beyond expected levels. The expansion criteria are defined in Section 5.

NUREG-1800 and NUREG-1801 provide general guidance for selecting and justifying condition monitoring inspection populations and sample sizes (i.e., consistent with Branch Technical Position RLSB-1 [46]), and this guidance has been used to develop the requirements for examination coverage. In particular, for redundant systems or multiple connections, examination coverage sampling is based on existing requirements in accepted condition monitoring sampling programs, such as the ASME Code Section XI, Subsection IWB-2500 Examination Category B-J piping welds and the Chapter XI.M33 program description in NUREG-1801, Revision 2, for selective leaching. In the ASME Code Section XI, IWB-2500 case, the sample population of 25% of the total population has been determined to be adequate. For the Chapter XI.M33 program, a 20% sample or no more than 25 component items is considered a statistically significant sample size. This sample size is also considered appropriate for one-time inspection programs where monitoring AMPs XI.M2, XI.M30 and XI.M39 are credited and for which no aging effects have been observed or are occurring very slowly and do not affect the component's or structure's intended function during the period of extended operation based on prior operating data. This "program cannot be used for structures or components subjected to known age-related degradation mechanisms or when the environment in the period of extended operation is not expected to be equivalent to that in the prior 40 years. Periodic inspections should be proposed in these cases." For the components for which a sample strategy is identified in the notes to Tables 4-1 through 4-6 all aspects of the environment are expected to remain equivalent over the period of extended operation with the exception of the degree of irradiation embrittlement. This degradation effect, while affecting toughness and flaw tolerance, does not cause flaw initiation. Thus, the periodicity specified to perform a repeated sample, coupled with sample expansion requirements, constitute a robust technical basis for using a sampling strategy.

The requirements for Primary and Expansion components for B&W, CE, and Westinghouse plants, respectively, are listed in Tables 4-1 through 4-6. For example, the Primary and Expansion requirements for Westinghouse internals are listed in Tables 4-3 and 4-6. These tables contain columns describing the component; any particular applicability requirement for that component; the degradation effect to be detected; the examination method; the examination coverage; and any linkage between the Primary and Expansion components. The technical bases for the examination requirements are contained in the aging management strategy reports [13, 14].

There are no specified examinations where inadequate coverage is anticipated to be an issue. However if a utility determines that the examination coverage is questionable with respect to meeting the intent of the guidelines, the condition should be entered in the utility's corrective action program for disposition.

The term "accessible" as used in Tables 4-1 through 4-6 is defined as a component surface or volume for which an examination is specified in accordance with MRP-228 that can be examined with the technologies specified in MRP-228. This accessibility is consistent with current ASME Section XI practices.

In some cases, consistent implementation of the inspection requirements of the ASME B&PV Code Section XI or requirements established by the NSSS vendors or Issue Program, to address degradation that manifested itself during the current operational life of the PWR fleet is sufficient for management of the postulated age-related degradation. These guidelines credit the Existing Programs listed in Tables 4-7 to 4-9 for aging management of the listed components. It is the responsibility of the plant owner to assure that these programs are properly implemented and meet established ASME or industry standards. The continued implementation of this existing guidance has been determined to adequately manage the aging effects for these components.

Table 4-1
B&W Plants Primary Components

Primary Item	Applicability	Effect (Mechanism)	Expansion Link (Note 1)	Examination Method/Frequency (Note 1,2)	Examination Coverage
B1.Plenum Cover Assembly & Core Support Shield Assembly a. Plenum cover weldment rib pads b. Plenum cover support flange c. Plenum cover support ring d. CSS top flange	All plants	Loss of material and associated loss of core clamping pre-load (Wear)	None	One-time physical measurement no later than two refueling outages following entry into the period of extended operation. Subsequent visual (VT-3) examination prior to the end of each 10-year ISI interval.	Determination of differential height of top of plenum rib pads/plenum cover support ring location to reactor vessel seating surface, with plenum in reactor vessel. See Figure 4-1.
B2.Control Rod Guide Tube Assembly CRGT spacer castings	All plants	Cracking (TE), including the detection of fractured spacers or missing screws (Note 3)	B2.1.Vent valve bodies	Visual (VT-3) examination during the next 10-year ISI interval. Subsequent examination prior to the end of each 10-year ISI interval. (Note 10)	Accessible surfaces at each of the four screw locations (at every 90°) of 100% of the CRGT spacer castings. (limited accessibility) (Note 12) See Figure 4-5.
B3.Vent Valve Assembly a. Vent valve top retaining ring b. Vent valve bottom retaining ring	All plants	Cracking (TE), including the detection of surface irregularities, such as damaged or fractured material in the retaining ring flanged region, or missing material (Note 3)	None	Visual (VT-3) examination during the next 10-year ISI interval. Subsequent examination during each 10-year ISI interval. (Note 10)	100% of accessible surfaces. (limited accessibility) See Figure 4-10.
B4.Vent Valve Assembly Original locking devices (pressure plate, spring retainer, spring, U-cover)	Note 4	Loss of material from locking device (Wear associated with jack screw spring and pressure plate)	None	Visual (VT-3) examination during the next 10-year ISI interval. (Note 5) Subsequent examination during each 10-year ISI interval.	100% of accessible surfaces, including the overall valve symmetry in the mounting ring and the overall jack screw thread extension from the lower retaining ring threaded flange. See Figure 4-10.

Table 4-1 (continued)
B&W Plants Primary Components

Primary Item	Applicability	Effect (Mechanism)	Expansion Link (Note 1)	Examination Method/Frequency (Note 1,2)	Examination Coverage
B5.Vent Valve Assembly Original locking devices (key ring, pin)	Note 4	Cracking (TE), including the detection of reportable indications	None	Visual (VT-3) examination during the next 10-year ISI interval. (Note 5) Subsequent examination during each 10-year ISI interval.	100% of accessible surfaces, including the overall valve symmetry in the mounting ring and the overall jack screw thread extension from the lower retaining ring threaded flange. See Figure 4-10.
B6.Vent Valve Assembly Modified locking devices (bolt locking cup, jackscrew locking cup, bolted block)	Note 4	Cracking (SCC), including detection of fractured or missing locking cups and welds, or the bolted block	None	Visual (VT-3) examination during the next 10-year ISI interval. (Note 5) Subsequent examination during each 10-year ISI interval.	100% of accessible surfaces, including the overall valve symmetry in the mounting ring and the overall jack screw thread extension from the lower retaining ring threaded flange. See Figure 4-10.
B7.Core Support Shield Assembly Upper core barrel (UCB) bolts and their locking devices	All plants	Bolts: Cracking (SCC–all bolts, Fatigue–original bolts only) Locking Devices: Loss of material, damaged or distorted or missing locking devices (Wear or Fatigue damage by failed bolts)	B7.1.UTS bolts and B8.1.LTS studs/nuts or bolts and their locking devices B7.2.SSHT bolts and their locking devices (DB only)	Bolts: Volumetric (UT) examination during the next 10-year ISI interval. Locking Devices: Visual (VT-3) examination during the next 10-year ISI interval. Subsequent examination during each 10-year ISI interval unless an evaluation of the baseline results, submitted for NRC staff approval, justifies a longer interval between examinations.	100% of accessible bolts and their locking devices. (Note 6) See Figure 4-7.

Table 4-1 (continued)
B&W Plants Primary Components

Primary Item	Applicability	Effect (Mechanism)	Expansion Link (Note 1)	Examination Method/Frequency (Note 1,2)	Examination Coverage
B8.Core Barrel Assembly Lower core barrel (LCB) bolts and their locking devices	All plants	Bolts: Cracking (SCC, IC/ISR/Fatigue/ Wear) (Note 7) Locking Devices: Loss of material, damaged or distorted or missing locking devices (Wear or Fatigue damage by failed bolts)	B7.1.UTS bolts and B8.1.LTS studs/nuts or bolts and their locking devices B7.2.SSHT bolts and their locking devices (DB only)	Bolts: Volumetric (UT) examination during the next 10-year ISI interval. Locking Devices: Visual (VT-3) examination during the next 10-year ISI interval. Subsequent examination during each 10-year ISI interval unless an evaluation of the baseline results, submitted for NRC staff approval, justifies a longer interval between examinations.	100% of accessible bolts and their locking devices. (Note 6) See Figure 4-8(a).
B9.Core Barrel Assembly Baffle-to-former bolts	All plants	Cracking (IASCC, IE, IC/ISR/Fatigue/ Wear, Overload) (Notes 3, 7)	B9.1.Baffle-to-baffle bolts B9.2.Core barrel-to-former bolts	Volumetric (UT) examination during the next 10-year ISI interval. Subsequent examination during each 10-year ISI interval. (Note 11)	100% of accessible bolts. (Note 6) See Figure 4-2.
B10.Core Barrel Assembly Baffle plates	All plants	Cracking (IE), including the detection of readily detectable cracking (Note 3)	B10.1.Core barrel cylinder (including vertical and circumferential seam welds) B10.2.Former plates B10.3.Lower grid rib section	Visual (VT-3) examination during the next 10-year ISI interval. Subsequent examination during each 10-year ISI interval. (Note 10)	100% of the accessible surfaces within one inch around each flow and bolt hole. See Figure 4-2.

Table 4-1 (continued)
B&W Plants Primary Components

Primary Item	Applicability	Effect (Mechanism)	Expansion Link (Note 1)	Examination Method/Frequency (Note 1,2)	Examination Coverage
B11.Core Barrel Assembly Locking devices, including locking welds, of baffle-to-former bolts and internal baffle-to-baffle bolts	All plants	Locking Devices: Cracking (IASCC, IE), including the detection of missing, non-functional, or removed locking devices (Note 3) Locking Welds: Cracking (IE), including the detection of missing, non-functional, or removed locking device welds (Notes 3, 8)	B11.1.Locking devices, including locking welds, of the external baffle-to-baffle bolts and core barrel-to-former bolts (Note 8)	Visual (VT-3) examination during the next 10-year ISI interval. Subsequent examination during each 10-year ISI interval. (Note 10)	100% of accessible baffle-to-former and internal baffle-to-baffle bolt locking devices and locking welds. (Note 6) See Figure 4-2.
B12.Flow Distributor Assembly Flow distributor (FD) bolts and their locking devices	All plants	Bolts: Cracking (SCC, Fatigue) Locking Devices: Loss of material, damaged or distorted or missing locking devices (Wear or Fatigue damage by failed bolts)	B7.1.UTS bolts and B8.1.LTS studs/nuts or bolts and their locking devices B7.2.SSHT bolts and their locking devices (DB only)	Bolts: Volumetric (UT) examination during the next 10-year ISI interval. Locking Devices: Visual (VT-3) examination during the next 10-year ISI interval. Subsequent examination during each 10-year ISI interval unless an evaluation of the baseline results, submitted for NRC staff approval, justifies a longer interval between examinations.	100% of accessible bolts and their locking devices. (Note 6) See Figure 4-8(a).
B13.Lower Grid Assembly Alloy X-750 dowel-to-guide block welds	All plants (except DB) (Note 9)	Cracking (SCC), including the detection of separated or missing locking welds, or missing dowels	Alloy X-750 dowel locking welds to the B13.1.upper and B13.2.lower grid fuel assembly support pads	Visual (VT-3) examination during the next 10-year ISI interval. Subsequent examination during each 10-year ISI interval.	Accessible surfaces of 100% of the dowel-to-guide block welds. See Figure 4-4.

Table 4-1 (continued)
B&W Plants Primary Components

Primary Item	Applicability	Effect (Mechanism)	Expansion Link (Note 1)	Examination Method/Frequency (Note 1,2)	Examination Coverage
B14.Lower Grid Assembly Shock pad bolts and their locking devices	TMI-1	Bolts: Cracking (SCC, Fatigue) Locking Devices: Loss of material, damaged or distorted or missing locking devices (Wear or Fatigue damage by failed bolts)	B7.1.UTS bolts and B8.1.LTS bolts and their locking devices	Bolts: Volumetric (UT) examination during the next 10-year ISI interval. Locking Devices: Visual (VT-3) examination during the next 10-year ISI interval. Subsequent examination during each 10-year ISI interval unless an evaluation of the baseline results, submitted for NRC staff approval, justifies a longer interval between examinations.	100% of accessible bolts and their locking devices. (Note 6) See Figure 4-4.
B15.Incore Monitoring Instrumentation (IMI) Guide Tube Assembly IMI guide tube spiders IMI guide tube spider-to-lower grid rib section welds	All plants	Spiders: Cracking (TE/IE), including the detection of fractured or missing spider arms (Note 3) Spider Welds: Cracking (IE), including separation of spider arms from the lower grid rib section at the weld (Note 3)	<u>B15.1.Lower grid fuel assembly support pad component items</u>: pad, pad-to-rib section welds, Alloy X-750 dowel, cap screws, and their locking welds (Note: the pads, dowels, and cap screws are included because of IE of the welds)	Visual (VT-3) examination during the next 10-year ISI interval. Subsequent examination during each 10-year ISI interval.	Spiders: 100% of the accessible top surfaces and 100% of the accessible spider surfaces adjacent to the spider casting welds. Spider Welds: 100% of the accessible welds to the adjacent lower grid rib section. See Figures 4-3 and 4-6.

Notes to Table 4-1:

1. Examination acceptance criteria and expansion criteria for the B&W component items and welds are detailed in Table 5-1.
2. Initial examinations may be scheduled concurrently with the next 10-year ISI but shall be performed no later than the final outage of the current 10-year ISI interval at the time of entry into the period of extended operation, except where otherwise noted.
3. Loss of ductility and fracture toughness, which are induced by thermal aging or neutron irradiation embrittlement, are not directly monitored. These effects are indirectly managed by performing periodic visual examinations capable of detecting cracking in the component.
4. As of May 2014, TMI-1 and DB have been verified to have original vent valve assembly locking devices only, while ONS-1, ONS-2, ONS-3, and ANO-1 have both original and modified vent valve assembly locking devices installed.

5. In addition to the Primary vent valve assembly visual (VT-3) examinations identified in this table, the following testing and inspection requirements are currently performed and will continue to be performed by the B&W units:

    A verification of the operation of each vent valve shall also be performed through manual actuation of the valve. Verify that the valves are not stuck in the open position and that no abnormal degradation has occurred. Examine the valves for evidence of leakage between the valve disc and the valve body (i.e., flow lines across the sealing surface), cracking of lock welds and locking cups, jackscrews for proper position, and wear. The frequency is defined in each unit's technical specifications or in their pump and valve inservice test programs (see BAW-2248A, page 4.3 and Table 4-1).

6. A minimum of 75% of the total population (examined + unexamined), including coverage consistent with the Expansion criteria in Section 5.3 of this document, must be examined for inspection credit.
7. The primary aging degradation mechanisms for loss of joint tightness for this item are IC and ISR. Fatigue and wear, which can also lead to cracking, are secondary aging degradation mechanisms after significant stress relaxation and loss of preload has occurred due to ISR/IC. Bolt stress relaxation cannot readily be inspected by NDE. Only bolt cracking is inspected by UT inspection.
8. The aging degradation mechanism of IASCC is only applicable to the baffle-to-former bolt and internal baffle-to-baffle bolt locking devices, not the baffle-to-former bolt and internal baffle-to-baffle bolt locking device welds. There are no Expansion component items for the baffle-to-former bolt and internal baffle-to-baffle bolt locking device welds for IASCC.
9. For the alternate configuration at DB, the difference in configuration shall be addressed on a unit-specific basis.
10. The term "10-year ISI interval" is intended to mean the plant's existing schedule associated with removal of the core barrel, i.e., during the next scheduled 10-year ISI interval examination. As defined in the ASME B&PV Code Section XI, the first inspection interval is defined as 10 years following the initial start of plant commercial startup, and successive inspection intervals are defined as 10 years following the previous inspection interval. Each inspection interval may be extended by as much as 1 year and may be reduced without restriction, provided the examinations required for the interval have been completed. Successive intervals shall not extend more than 1 year beyond the original pattern of 10-year intervals and shall not exceed 11 years in length.
11. This assumes that all units operating as of December 2011 have performed baseline (initial) volumetric (UT) examinations no later than two refueling outages from the beginning of their first license renewal period.
12. At one of the CRGT locations at ANO-1, a device was installed to monitor water level that required removal of the center control rod drive mechanism and installation of an assembly that blocks access to all of the CRGT spacer castings except the top spacer casting. Therefore, for all plants except ANO-1, 100% of the CRGT spacer castings refers to 690 CRGT spacer castings. There is no requirement to examine this set of CRGT spacer castings at this one CRGT assembly at ANO-1, and 100% is defined as 680 CRGT spacer castings for ANO-1.

Table 4-2
CE Plants Primary Components

Primary Item	Applicability	Effect (Mechanism)	Expansion Link (Note 1)	Examination Method/Frequency (Note 1)	Examination Coverage
C1.Core Shroud Assembly (Bolted) Core shroud bolts	Bolted plant designs	Cracking (IASCC, Fatigue) Aging Management (IE and ISR)	C1.1.Core support column bolts, C1.2.Barrel-shroud bolts	Baseline volumetric (UT) examination between 25 and 35 EFPY, with subsequent examination on a ten-year interval.	100% of accessible bolts (see Note 2). See Figure 4-25.
C2.Core Shroud Assembly (Welded) Core shroud plate-former plate weld	Plant designs with core shrouds assembled in two vertical sections	Cracking (IASCC) Aging Management (IE)	C2.1.Remaining axial welds	Enhanced visual (EVT-1) examination no later than 2 refueling outages from the beginning of the license renewal period and subsequent examination on a ten-year interval.	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 central flange and horizontal stiffeners and ¾" of adjacent base metal. See Figure 4-26.
C3.Core Shroud Assembly (Welded) Shroud plates	Only plants with welded core shrouds assembled with full-height shroud plates	Cracking (IASCC) Aging Management (IE)	C3.1.Remaining axial welds, C3.2.Ribs and rings	Enhanced visual (EVT-1) examination no later than 2 refueling outages from the beginning of the license renewal period and subsequent examination on a ten-year interval.	Axial weld seams at the core shroud re-entrant corners, at the core mid-plane (± three feet in height) as visible from the core side of the shroud and ¾" of adjacent base metal. See Figure 4-27.

Table 4-2 (continued)
CE Plants Primary Components

Primary Item	Applicability	Effect (Mechanism)	Expansion Link (Note 1)	Examination Method/Frequency (Note 1)	Examination Coverage
C4.Core Shroud Assembly (Bolted) Assembly	Bolted plant designs	Distortion (Void Swelling), including: • Abnormal interaction with fuel assemblies • Gaps along shroud plate joints • Relative Vertical displacement of shroud plates Aging Management (IE)	None	Visual (VT-3) examination no later than 2 refueling outages from the beginning of the license renewal period. Subsequent examinations on a ten-year interval.	Core side surfaces: High fluence shroud plate joints Top and bottom alignment of shroud plates Bolts and locking devices See Figure 4-28.
C4a.Core Shroud Assembly (Welded) Assembly	Plant designs with core shrouds assembled in two vertical sections	Distortion (Void Swelling), as evidenced by measurable separation between the upper and lower core shroud segments Aging Management (IE)	None	Visual (VT-1) examination no later than 2 refueling outages from the beginning of the license renewal period. Subsequent examinations on a ten-year interval.	100% of the horizontal seam between the upper and lower core shroud segments. (Note 3) See Figure 4-26.
C5.Core Support Barrel Assembly Upper flange weld (UFW)	All plants	Cracking (SCC)	C5.2.Upper Girth Weld (UGW), C5.1.Lower Girth/Flange Weld (LGW/LFW), C5.3.Upper Axial Welds (UAW), C5.4.Lower Core Support Beams	Enhanced visual (EVT-1) examination no later than 2 refueling outages from the beginning of the license renewal period. Subsequent examinations on a ten-year interval.	100% of the accessible weld length of one side of the UFW and ¾″ of adjacent base metal shall be examined. (Note 4) See Figure 4-29.
C6.Core Support Barrel Assembly Middle Girth Weld (MGW)	All plants	Cracking (SCC, IASCC) Aging Management (IE)	C6.1.Middle Axial Weld (MAW), C6.2.Lower Axial Weld (LAW), C6.3.Core Support Columns	Enhanced visual (EVT-1) examination no later than 2 refueling outages from the beginning of the license renewal period. Subsequent examinations on a ten-year interval.	100% of the accessible weld length of the OD of the MGW and ¾″ of adjacent base metal shall be examined. (Note 4) See Figure 4-29.

Table 4-2 (continued)
CE Plants Primary Components

Primary Item	Applicability	Effect (Mechanism)	Expansion Link (Note 1)	Examination Method/Frequency (Note 1)	Examination Coverage
C7.Core Support Barrel Assembly CSB Flexure Weld (CSBFW)	All plants with welded core shrouds	Cracking (Fatigue, SCC)	None	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.	Examination coverage to be defined by evaluation to determine the potential location and extent of cracking. See Figure 4-30.
C9.Lower Support Structure Core support plate	All plants with a core support plate	Cracking (Fatigue) Aging Management (IE)	None	If screening for fatigue cannot be satisfied by plant-specific evaluation, 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.	Examination coverage to be defined by evaluation to determine the potential location and extent of fatigue cracking. See Figure 4-28.

Table 4-2 (continued)
CE Plants Primary Components

Primary Item	Applicability	Effect (Mechanism)	Expansion Link (Note 1)	Examination Method/Frequency (Note 1)	Examination Coverage
C10.Upper Internals Assembly Fuel alignment plate	Only plants with welded core shrouds assembled with full-height shroud plates	Cracking (Fatigue), Aging management (IE)	None	If screening for fatigue cannot be satisfied by plant-specific evaluation, 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.	Examination coverage to be defined by evaluation to determine the potential location and extent of fatigue cracking. See Figure 4-31.
C11.Control Element Assembly Instrument guide tubes	All plants with instrument guide tubes attached to the CEA shroud assemblies	Cracking (SCC, Fatigue) that results in missing supports or separation at the welded joint between the tubes and supports	C11.1.Remaining instrument guide tubes within the CEA shroud assemblies	Visual (VT-3) examination, no later than 2 refueling outages from the beginning of the license renewal period. Subsequent examination on a ten-year interval.	100% of instrument guide tubes in peripheral CEA shroud assemblies (i.e., those adjacent to the perimeter of the fuel alignment plate), focusing on the supports and the connecting welds between the supports and the guide tube and CEA shroud. See Figure 4-32.
C12.Lower Support Structure Deep beams	Only plants with welded core shrouds assembled with full-height shroud plates	Cracking (Fatigue) that results in a detectable surface-breaking indication in the welds or beams. Aging Management (IE)	Remaining deep beams	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 adequacy of remaining fatigue life cannot be demonstrated.	Examine 25% of the total number of beam-to-beam welds. Coverage on each weld includes the top four inches of the weld in the vertical orientation and ¾" of adjacent base metal. The inspection coverage must be evenly distributed across the deep beam structure. (Notes 5, 6, and 7) See Figure 4-33.

Notes to Table 4-2:

1. Examination acceptance criteria and expansion criteria for the CE components are in Table 5-2.
2. A minimum of 75% of the total core shroud bolts population (examined + unexamined), including coverage consistent with the Expansion criteria in Table 5-2, must be examined for inspection credit.
3. If evidence of distortion is detected by visual exam, consideration should be given to making supplementary measurements (minimum of 3 to 5, depending on extent of observed condition) of gap opening between the upper and lower core shroud segments.

4. Examination coverage requires a minimum of 75% of the length of either the ID or the OD of the weld being examined.
5. The stated coverage requirement is the minimum if no significant indications are found. However the Examination Acceptance criteria in Section 5 require that additional coverage must be achieved in the same outage if significant flaws are found. This contingency should be considered for inspection planning purposes.
6. Inspections subsequent to the first 25% sampling inspection must cover weld length that was not inspected during previous examinations. A minimum of 20% out of the 25% inspection coverage of each subsequent inspection must include previously un-inspected weld length as possible within the limitation of the remaining accessible un-inspected weld.
7. Justification that adequate distribution of the inspection coverage has been achieved can be based on geometric or layout arguments. For example, since each rectangular axial compartment in the structure has four welds around it, which are shared with the adjacent compartments on two sides, the inspection coverage could consist of the selection of one weld from every second compartment in the structure.

Table 4-3
Westinghouse Plants Primary Components

Primary Item	Applicability	Effect (Mechanism)	Expansion Link (Note 1)	Examination Method/Frequency (Note 1)	Examination Coverage
W1.Control Rod Guide Tube Assembly Guide plates (cards)	All plants (See WEC NSAL-17-1)	Loss of Material (Wear)	None	Per the requirements of WCAP-17451-P, including subsequent examinations (Note 5).	Examination coverage per the requirements of WCAP-17451-P, Revision 1 (Note 5) See Figure 4-11
W2.Control Rod Guide Tube Assembly Lower flange welds	All plants	Cracking (SCC, Fatigue) Aging Management (IE and TE)	W2.1.Remaining CRGT assembly lower flange welds W2.2.BMI column bodies	Enhanced visual (EVT-1) examination to determine the presence of crack-like surface flaws in flange welds no later than 2 refueling outages from the beginning of the license renewal period and subsequent examination on a ten-year interval.	100% of outer (accessible) CRGT lower flange weld surfaces and 0.25-inch of the adjacent base metal on the individual periphery CRGT assemblies. (Note 2) See Figure 4-12.
W3.Core Barrel Assembly Upper flange Weld (UFW)	All plants	Cracking (SCC)	W3.1.Upper girth weld (UGW), W3.3.lower flange weld (LFW), W3.2.Upper axial welds (UAW), and W3.4.Lower support forging or casting	Enhanced visual (EVT-1) examination, no later than 2 refueling outages from the beginning of the license renewal period and subsequent examination on a ten-year interval.	100% of the accessible weld length of one side of the UFW and ¾" of adjacent base metal shall be examined. (Note 6) See Figure 4-13.
W4.Core Barrel Assembly Lower girth weld (LGW)	All plants	Cracking (SCC, IASCC), Aging Management (IE)	W4.1.Upper core plate, W4.4.Lower support column bodies (cast, non-cast), W4.2.Middle axial welds (MAW), W4.3.Lower axial welds (LAW)	Enhanced visual (EVT-1) examination, no later than 2 refueling outages from the beginning of the license renewal period and subsequent examination on a ten-year interval.	100% of the accessible weld length of the OD of the LGW and ¾" of adjacent base metal shall be examined. (Note 6) See Figure 4-13.
W5.Baffle-Former Assembly Baffle-edge bolts	All plants with baffle-edge bolts	Cracking (IASCC, Fatigue) that results in ● Lost or broken locking devices ● Failed or missing bolts ● Protrusion of bolt heads Aging Management (IE and ISR) (Note 4)	None	Visual (VT-3) examination, with baseline examination between 20 and 40 EFPY and subsequent examinations on a ten-year interval.	Bolts and locking devices on high fluence seams. 100% of components accessible from core side. See Figure 4-14.

Table 4-3 (continued)
Westinghouse Plants Primary Components

Primary Item	Applicability	Effect (Mechanism)	Expansion Link (Note 1)	Examination Method/Frequency (Note 1)	Examination Coverage
W6.Baffle-Former Assembly Baffle-former bolts (Note 7)	All plants (See WEC NSAL-16-1)	Cracking (IASCC, Fatigue) Aging Management (IE and ISR) (Note 4)	W6.2.Lower support column bolts, W6.1.Barrel-former bolts	Baseline volumetric (UT) examination interval is dependent on the plant design (Note 8). Subsequent examination is dependent on the plant design and the results of the baseline inspection (Note 9).	100% of accessible bolts. (Note 3) See Figure 4-15.
W7.Baffle-Former Assembly Assembly (Includes: Baffle plates, baffle edge bolts, corner bolts, and indirect effects of void swelling in former plates)	All plants	Distortion (Void Swelling), or Cracking (IASCC) that results in • Abnormal interaction with fuel assemblies • Gaps between plates • Vertical displacement of baffle plates • Broken or damaged edge bolts	None	Visual (VT-3) examination to check for evidence of distortion, with baseline examination between 20 and 40 EFPY and subsequent examinations on a ten-year interval.	Core side surface: • High fluence baffle joints • Top and bottom edge of baffle plates • Bolts and locking devices See Figure 4-16.
W8.Alignment and Interfacing Components Internals hold down spring	All plants with 304 stainless steel hold down springs	Distortion (Loss of Load due to Stress Relaxation)	None	Direct measurement of spring height within three cycles of the beginning of (before or after) the license renewal period. If the first set of measurements is not sufficient to assess remaining life, additional spring height measurements will be required.	Measurements should be taken at several points around the circumference of the spring, with a statistically adequate number of measurements at each point to minimize uncertainty. See Figure 4-17.
W9.Thermal Shield Assembly Thermal shield flexures	All plants with thermal shields (See WEC TB-19-5)	Cracking (Fatigue) or Loss of Material (Wear) that results in thermal shield flexures excessive wear, fracture, or complete separation	None	Visual (VT-3) no later than 2 refueling outages from the beginning of the license renewal period. Subsequent examinations on a ten-year interval.	100% of accessible surfaces of 100% of thermal shield flexures. (Note 10) See Figure 4-18.

Notes to Table 4-3:

1. Examination acceptance criteria and expansion criteria for the Westinghouse components are in Table 5-3.
2. A minimum of 75% of the total identified sample population must be examined.
3. A minimum of 75% of the total bolt population (examined + unexamined), including coverage consistent with the Expansion criteria in Table 5-3, must be examined for inspection credit.
4. Void swelling effects on this component are managed through management of void swelling on the entire baffle-former assembly.

5. In WCAP-17451-P the baseline examination schedule has been adjusted for various CRGT designs, the extent of individual CRGT examination modified, and flexible subsequent examination regimens correlating to initial baseline sample size, accuracy of wear estimation, and examination results. Initial inspection prior to the license renewal period may be required. Use WCAP-17451-P [37], Revision 1, including the modified requirements due to the interim guidance provided in EPRI letter MRP 2018-007 dated 3/7/2018 [47] and PWROG letter OG-18-46 dated 2/20/2018 [38].
6. Examination coverage requires a minimum of 50% of the length of either the ID or the OD of the weld being examined.
7. Baffle-former bolt inspection includes inspection of the corner plate bolts when applicable.
8. In accordance with MRP 2017-009 [39] and MRP 2017-010 [41], Tier 1 plants are to perform the baseline UT examination by 20 EFPY or during the next refueling outage after March 1, 2016. Per MRP 2017-009 [39], Tier 2 plants are to perform the baseline UT examination at no later than 30 EFPY (initial Tier 2 plant baseline UT exams performed prior to 1/1/2018 are acceptable). All other remaining plants are to perform the baseline UT examination at no later than 35 EFPY.
9. Re-examination periods shall be determined by plant-specific evaluation per the MRP-227 Needed Requirement 7.5 as documented and dispositioned in the owner's plant corrective action program. If atypical or aggressive baffle-former bolt degradation as defined in MRP 2017-009 [39] (i.e., ≥3% of baffle-former bolts with UT or visual indications or clustering* for downflow plants and ≥5% of baffle-former bolts with UT or visual indications or clustering* for upflow plants) is observed, the interim guidance (MRP 2016-021 [40] and MRP 2017-009 [39]) provides limitations to the permitted reinspection interval (not to exceed 6 years maximum) unless further evaluation is performed to justify a longer interval (See Applicant/Licensee Action Item 1 in the NRC SE for evaluation submittal requirements [35]). If evaluation justifies a longer reinspection interval, it is not permitted to exceed 10 years.

   *"Clustering" is defined per NSAL-16-1 Rev. 1 [42] as three or more adjacent defective baffle-former bolts or more than 40% defective baffle-former bolts on the same baffle plate. Untestable bolts should be reviewed on a plant-specific basis consistent with WCAP-17096-NP-A for determination if these should be considered when evaluating clustering.
10. See Westinghouse Technical Bulletin TB-19-5 dated 10/9/2019 and MRP 2019-017 dated 5/31/2019 for additional details on inspection recommendations.

Table 4-4
B&W Plants Expansion Components

Expansion Item	Applicability	Effect (Mechanism)	Primary Link (Note 1)	Examination Method/Frequency (Note 1)	Examination Coverage
B13.1.**Upper Grid Assembly** Alloy X-750 dowel-to-upper grid fuel assembly support pad welds	All plants (Note 2)	Cracking (SCC), including the detection of separated or missing locking welds, or missing dowels	B13.Alloy X-750 dowel-to-guide block welds	Visual (VT-3) examination. Subsequent examination during each 10-year interval unless an applicant/licensee provides an evaluation for NRC staff approval that justifies a longer interval between inspections.	Accessible surfaces of 100% of the support pad dowel locking welds. See Figure 4-6. (These are similar to the lower grid fuel assembly support pads)
B2.1.**Vent Valve Assembly** Vent valve bodies (Note 9)	All plants	Cracking (TE), including the detection of fractured vent valve bodies, surface irregularities (damaged, grossly cracked, or missing portions) of the vent valve bodies, as a result of damage to the jackscrews or locking devices (Note 3)	B2.CRGT spacer castings	Visual (VT-3) examination. Subsequent examination during each 10-year interval unless an applicant/licensee provides an evaluation for NRC staff approval that justifies a longer interval between inspections.	Accessible surfaces of 100% of the vent valve bodies on the plenum side (core side) of the vent valve. (Limited accessibility) See Figures 4-9 and 4-10. (Note 4)
B7.1.**Core Barrel Assembly** Upper thermal shield (UTS) bolts and their locking devices	All plants	Bolts: Cracking (SCC—UTS and SSHT bolts, IC/ISR/Wear/Fatigue–SSHT bolts) (Note 7) Locking Devices: Loss of material, damaged, distorted or missing locking devices (Wear or Fatigue damage by failed bolts)	B7.UCB, B8.LCB, or B12.FD bolts and their locking devices B14.Shock pad bolts and their locking devices (TMI-1 only)	Bolts: Volumetric (UT) examination. Locking Devices: Visual (VT-3) examination. Subsequent examination during each 10-year interval unless an applicant/licensee provides an evaluation for NRC staff approval that justifies a longer interval between inspections.	100% of accessible bolts and their locking devices. (Note 5) See Figure 4-7 (for UTS bolts) and Figure 4-8(b) (for SSHT bolts).
B7.2.**Core Barrel Assembly** Surveillance specimen holder tube (SSHT) bolts and their locking devices	DB				
Core Barrel Assembly B10.1.Core barrel cylinder (including vertical and center circumferential seam welds) B10.2.Former plates	All plants	Cracking (IE) (Note 3)	B10.Baffle plates	No examination requirements. Justify by evaluation or by replacement.	Inaccessible. See Figure 4-2.

Table 4-4 (continued)
B&W Plants Expansion Components

Expansion Item	Applicability	Effect (Mechanism)	Primary Link (Note 1)	Examination Method/Frequency (Note 1)	Examination Coverage
Core Barrel Assembly B9.1.Baffle-to-baffle bolts B9.2.Core barrel-to-former bolts	All plants	Cracking (IASCC, IE, IC/ISR/Fatigue/Wear, Overload) (Notes 3, 6, 8)	B9.Baffle-to-former bolts	Internal baffle-to-baffle bolts: No examination requirements. Justify by evaluation or by replacement.	An acceptable examination technique currently not available. See Figure 4-2.
				External baffle-to-baffle bolts, Core barrel-to-former bolts: No examination requirements. Justify by evaluation or by replacement.	Inaccessible. See Figure 4-2.
Core Barrel Assembly B11.1.Locking devices, including locking welds, of the external baffle-to-baffle bolts and core barrel-to-former bolts	All plants	Cracking (IE) (Note 3)	B11.Locking devices, including locking welds, of baffle-to-former bolts and internal baffle-to-baffle bolts	No examination requirements. Justify by evaluation or by replacement.	Inaccessible. See Figure 4-2.
B15.1.**Lower Grid Assembly** <u>Lower grid fuel assembly support pad component items</u>: pad, pad-to-rib section welds, Alloy X-750 dowel, cap screws, and their locking welds (Note: the pads, dowels, and cap screws are included because of IE of the welds)	All plants	Cracking (IE), including the detection of separated or missing welds, missing support pads, dowels, cap screws and locking welds, or misalignment of the support pads (Note 3)	B15.IMI guide tube spiders and spider-to-lower grid rib section welds	Visual (VT-3) examination. Subsequent examination during each 10-year interval unless an applicant/licensee provides an evaluation for NRC staff approval that justifies a longer interval between inspections.	Accessible surfaces of the pads, dowels, cap screws, and associated welds for 100% of the lower grid fuel assembly support pads. See Figure 4-6.
Lower Grid Assembly B10.3.Lower grid rib section	All plants	Cracking (IE), including the detection of readily detectable cracking (Note 3)	B10.Baffle plates	Visual (VT-3) examination. Subsequent examination during each 10-year interval unless an applicant/licensee provides an evaluation for NRC staff approval that justifies a longer interval between inspections.	100% of accessible surfaces of the lower grid rib section heat-affected zone (HAZ) adjacent to the spider-to-lower grid rib section welds. See Figures 4-3 and 4-6.

Table 4-4 (continued)
B&W Plants Expansion Components

Expansion Item	Applicability	Effect (Mechanism)	Primary Link (Note 1)	Examination Method/Frequency (Note 1)	Examination Coverage
B13.2.**Lower Grid Assembly** Alloy X-750 dowel-to-lower grid fuel assembly support pad welds	All plants (Note 2)	Cracking (SCC), including the detection of separated or missing locking welds, or missing dowels	B13.Alloy X-750 dowel-to-guide block welds	Visual (VT-3) examination. Subsequent examination during each 10-year interval unless an applicant/licensee provides an evaluation for NRC staff approval that justifies a longer interval between inspections.	Accessible surfaces of 100% of the support pad dowel locking welds. See Figure 4-6.
B8.1.**Lower Grid Assembly** Lower thermal shield (LTS) bolts (ANO-1, DB, and TMI-1) or studs/nuts (ONS-1, ONS-2, and ONS-3) and their locking devices	All plants	Bolts or Studs/Nuts: Cracking (SCC) Locking Devices: Loss of material, damaged, distorted or missing locking devices (Wear or Fatigue damage by failed bolts or studs/nuts)	B7.UCB, B8.LCB, or B12.FD bolts and their locking devices B14. Shock pad bolts and their locking devices (TMI-1 only)	Bolts or Studs/Nuts: Volumetric (UT) examination. Locking Devices: Visual (VT-3) examination. Subsequent examination during each 10-year interval unless an applicant/licensee provides an evaluation for NRC staff approval that justifies a longer interval between inspections.	100% of accessible bolts or studs/nuts and their locking devices. (Note 5) See Figure 4-8.

Notes to Table 4-4:

1. Examination acceptance criteria and expansion criteria for the B&W component items and welds are in Sections 5.1 - 5.3 of this document. Degradation exhibited in any of these primary links requires expansion in accordance with the criteria noted in Table 5-1.
2. For the alternate configuration at DB, the difference in configuration shall be addressed on a unit-specific basis.
3. Loss of ductility and fracture toughness, which are induced by thermal aging or neutron irradiation embrittlement, are not directly monitored. These effects are indirectly managed by performing periodic visual examinations capable of detecting cracking in the component.
4. It is known that some of the vent valves originally-installed to the B&W units were replaced with spares due to locking device issues. While the ferrite contents of the originally-installed vent valve bodies are known, the serial number [S/N] and corresponding ferrite contents of the currently-installed vent valve bodies is not fully known for the B&W units. Therefore, each utility shall identify the body S/N and heat number for each installed vent valve to determine installed heat of material and the ferrite content. The S/N numbers may be visible with an underwater video camera, e.g., during vent valve exercising. If the ferrite content of the currently-installed vent valve bodies can be verified to be under the TE screening criterion for the material, the vent valve bodies can be removed as an Expansion component item for the particular B&W unit and become a No Additional Measures component item. However, since vent valve assemblies are typically replaced in whole rather than by part, each time any changes are made to vent valve assemblies (e.g., replacement of a whole assembly), the vent valve body serial number, heat number, and ferrite content need to be verified. If, at any time, the ferrite content of any installed vent valve body is greater than the screening criterion for thermal embrittlement, that vent valve body will need to be included as an Expansion component item.
5. A minimum of 75% of the total population (examined + unexamined) must be examined for inspection credit.

6. The primary aging degradation mechanisms for loss of joint tightness for this item are IC and ISR. Fatigue and wear, which can also lead to cracking, are secondary aging degradation mechanisms after significant stress relaxation and loss of preload has occurred due to IC/ISR. Bolt stress relaxation cannot readily be inspected by NDE. Only bolt cracking could be inspected by UT inspection if it were possible for these bolts. Therefore, the effects of loss of joint tightness and/or cracking on the functionality of these bolts relative to the entire core barrel assembly will be addressed by analysis of the core barrel assembly.
7. This table entry for the SSHT bolts also includes the aging degradation mechanisms of ISR/IC/wear/fatigue for the compression collar and washer of the SSHT bolt.
8. The aging degradation mechanism consequence of overload is only applicable to the internal baffle-to-baffle bolts, not the core barrel-to-former bolts and external baffle-to-baffle bolts.
9. As noted in Section 3.1.3.1 of the SE for MRP-227, Rev. 1, licensees have the option of screening out the vent valve bodies using certified material test report (CMTR) data, or using the generic technical report PWROG-15032-NP [34] to show that the vent valve bodies have a high probability of having ferrite below the screening criterion for TE.

Table 4-5
CE Plants Expansion Components

Expansion Item	Applicability	Effect (Mechanism)	Primary Link (Note 1)	Examination Method/Frequency (Note 1)	Examination Coverage
Core Shroud Assembly (Bolted) C1.2.Barrel-shroud bolts	Bolted plant designs	Cracking (IASCC, Fatigue) Aging Management (IE and ISR)	C1.Core shroud bolts	Volumetric (UT) examination. Reinspection every 10 years following initial inspection.	100% of accessible barrel-shroud bolts or as supported by plant-specific justification. See Figure 4-34.
Core Support Barrel Assembly C5.1.Lower Girth Weld (LGW)	All plants	Cracking (SCC, Fatigue)	C5.Upper Flange Weld (UFW)	Enhanced visual (EVT-1) examination. Reinspection every 10 years following initial inspection.	100% of the accessible weld length of the OD of the LGW and ¾" of adjacent base metal shall be examined. (Note 5) See Figures 4-29 and 4-30.
Core Support Barrel Assembly C5.2.Upper Girth Weld (UGW) and C5.3.Upper Axial Weld (UAW)	All plants	Cracking (SCC) Aging Management (IE)	C5.Upper Flange Weld (UFW)	Enhanced visual (EVT-1) examination. Reinspection every 10 years following initial inspection.	100% of the accessible weld length of one side of the UGW and UAW and ¾" adjacent base metal shall be examined. (Notes 2 and 5) See Figure 4-29.
Lower Support Structure C5.4.Lower core support beams	All plants except those with welded core shrouds assembled with full-height shroud plates	Cracking (SCC, Fatigue) including damaged or fractured material	C5.Upper Flange Weld (UFW)	Enhanced visual (EVT-1) examination. Reinspection every 10 years following initial inspection.	25% of welds and ¾" of adjacent base metal or locations as justified by plant-specific evaluation. See Figure 4-35.
Core Support Barrel Assembly C6.1.Middle Axial Weld (MAW), C6.2.Lower Axial Weld (LAW)	All plants	Cracking (SCC, IASCC) Aging Management (IE)	C6.Middle Girth Weld (MGW)	Enhanced visual (EVT-1) examination Reinspection every 10 years following initial inspection.	100% of the accessible weld length of the OD of the MAW and LAW and ¾" of adjacent base metal shall be examined. (Note 5) See Figure 4-29.

Table 4-5 (continued)
CE Plants Expansion Components

Expansion Item	Applicability	Effect (Mechanism)	Primary Link (Note 1)	Examination Method/Frequency (Note 1)	Examination Coverage
Core Shroud Assembly (Bolted) C1.1.Core support column bolts	Bolted plant designs	Cracking (IASCC, Fatigue) Aging Management (IE)	C1.Core shroud bolts	Ultrasonic (UT) examination. Reinspection every 10 years following initial inspection.	100% (or as supported by plant-specific analysis) of core support column bolts (Minimum of 75% of the total population). See Figure 4-36.
Core Shroud Assembly (Welded) C2.1.Remaining axial welds	Plant designs with core shrouds assembled in two vertical sections	Cracking (IASCC), Aging Management (IE)	C2.Core shroud plate-former plate weld	Enhanced visual (EVT-1) examination. Reinspection every 10 years following initial inspection.	75% of the remaining axial weld length and ¾″ of adjacent base metal as visible from the core side of the shroud other than that already inspected under the primary link. See Figure 4-26.
Core Shroud Assembly (Welded) C3.1.Remaining axial welds, C3.2.Ribs and rings	Only plants with welded core shrouds assembled with full-height shroud plates	Cracking (IASCC), Aging Management (IE)	C3.Shroud plates of welded core shroud assemblies	Remaining axial welds: Enhanced visual (EVT-1) examination. Reinspection every 10 years following initial inspection. Ribs and rings: No examination requirements. Justify by evaluation or by replacement.	Remaining axial welds: 75% of the remaining axial weld length and ¾″ of adjacent base metal as visible from the core side of the shroud other than that already inspected under the primary link. Ribs and rings: Inaccessible See Figure 4-37.
Control Element Assembly C11.1.Remaining instrument guide tubes	All plants with instrument guide tubes in the CEA shroud assembly	Cracking (SCC, Fatigue) that results in missing supports or separation at the welded joint between the tubes and supports.	C11.Peripheral instrument guide tubes within the CEA shroud assemblies	Visual (VT-3) examination. Reinspection every 10 years following initial inspection.	100% of tubes in CEA shroud assemblies (minimum of 75% of the total population of remaining instrument guide tubes) focusing on the supports and the connecting welds between the supports and the guide tube and CEA shroud. See Figure 4-32.

Table 4-5 (continued)
CE Plants Expansion Components

Expansion Item	Applicability	Effect (Mechanism)	Primary Link (Note 1)	Examination Method/Frequency (Note 1)	Examination Coverage
Lower Support Structure C6.3.Core support columns	Plants with full-height bolted or half-height welded core shroud plates	Cracking (SCC, IASCC, Fatigue including damaged or fractured material) Aging Management (IE, TE)	C6.Middle Girth Weld (MGW)	Visual (VT-3) examination. Reinspection every 10 years following initial inspection.	Plants with full-height bolted core shroud plates: 25% of the total number of column assemblies (both visible and non-visible from above the core support plate) using a VT-3 examination from above the core support plate. The inspection coverage must be evenly distributed across the population of column assemblies. Plants with core shrouds assembled in two vertical sections: 25% of the accessible surfaces of the core support column welds, from the top side of the core support plate. The inspection coverage must be evenly distributed across the population of core support column welds. (Notes 3 and 4) See Figure 4-36.

Notes to Table 4-5:

1. Examination acceptance criteria and expansion criteria for the CE components are in Table 5-2.
2. Examination coverage requires examination of either the ID or the OD of the weld.
3. The stated minimum coverage requirement is the minimum if no significant indications are found. However, the Examination Acceptance criteria in Section 5 require that additional coverage must be achieved in the same outage if significant flaws are found. This contingency should be considered for inspection planning purposes.
4. Justification that adequate distribution of the inspection coverage has been achieved can be based on geometric or layout arguments. Possible examples include, but are not limited to, inspection of all column assemblies or accessible core support column welds in one quadrant of the core support plate (based on the azimuthal symmetry of the plate) or inspecting every fourth column or weld across the entire plate.
5. Examination coverage requires a minimum of 75% of the weld length for either the ID or the OD of the weld being examined.

Table 4-6
Westinghouse Plants Expansion Components

Expansion Item	Applicability	Effect (Mechanism)	Primary Link (Note 1)	Examination Method/Frequency (Note 1)	Examination Coverage
Control Rod Guide Tube Assembly W2.1.Remaining CRGT lower flange welds	All plants	Cracking (SCC, Fatigue) Aging Management (IE and TE)	W2.CRGT Lower Flange Welds	Enhanced visual (EVT-1) examination to determine the presence of crack-like surface flaws in flange welds. Subsequent examination on a ten-year interval.	A minimum of 75% of the CRGT assembly lower flange weld surfaces and 0.25-inch of the adjacent base metal for the flange welds not inspected under the primary link.
Bottom Mounted Instrumentation System W2.2.Bottom-mounted instrumentation (BMI) column bodies	All plants	Cracking (Fatigue) including the detection of completely fractured column bodies Aging Management (IE)	W2.CRGT Lower Flange Welds	Visual (VT-3) examination. Reinspection every 10 years following initial inspection.	100% of BMI column bodies for which difficulty is detected during flux thimble insertion/withdrawal. See Figure 4-24.
Core Barrel Assembly W3.1.Upper Girth Weld (UGW)	All plants	Cracking (SCC)	W3.Upper Core Barrel Flange Weld (UFW)	Enhanced visual (EVT-1) examination. Reinspection every 10 years following initial inspection.	100% of the accessible weld length of one side of the UGW and ¾″ of adjacent base metal shall be examined (Notes 2 and 5). See Figure 4-13.
Core Barrel Assembly W3.2.Upper Axial Weld (UAW)	All plants	Cracking (SCC)	W3.Upper Core Barrel Flange Weld (UFW)	Enhanced visual (EVT-1) examination. Reinspection every 10 years following initial inspection.	100% of the accessible weld length of one side of the UAW and ¾″ of adjacent base metal shall be examined (Notes 2 and 5). See Figure 4-13.
Core Barrel Assembly W3.3.Lower Flange Weld (LFW)	All plants	Cracking (SCC)	W3.Upper Core Barrel Flange Weld (UFW)	Enhanced visual (EVT-1) examination. Reinspection every 10 years following initial inspection.	100% of the accessible weld length of the OD surface of the LFW and ¾″ of adjacent base metal shall be examined (Note 5). See Figure 4-13.

Table 4-6 (continued)
Westinghouse Plants Expansion Components

Expansion Item	Applicability	Effect (Mechanism)	Primary Link (Note 1)	Examination Method/Frequency (Note 1)	Examination Coverage
Lower Internals Assembly W3.4.Lower support forging or castings	All plants	Cracking (SCC) Aging Management (TE in Casting) Ref.[34]	W3.Upper Core Barrel Flange Weld (UFW)	Visual (VT-3) examination. Reinspection every 10 years following initial inspection.	Minimum of 25% of bottom (non-core side) surface (Note 3). See Figure 4-20.
Upper Internals Assembly W4.1.Upper core plate	All plants	Cracking (Fatigue), Wear, Aging Management (IE)	W4.Lower Girth Weld (LGW)	Visual (VT-3) examination. Reinspection every 10 years following initial inspection.	Minimum of 25% of core side surfaces (Note 3). See Figure 4-19.
Core Barrel Assembly W4.2.Middle Axial Welds (MAW) and W4.3.Lower Axial Welds (LAW)	All plants	Cracking (SCC, IASCC) Aging Management (IE)	W4.Lower Girth Weld (LGW)	Enhanced visual (EVT-1) examination. Reinspection every 10 years following initial inspection	100% of the accessible weld length of the OD of the MAW and LAW and ¾" of adjacent base metal shall be examined (Notes 5 and 6). See Figure 4-13.
Lower Support Assembly W4.4.Lower support column bodies (both cast and non-cast)	All plants	Cracking (IASCC) Aging Management (IE)	W4.Lower Girth Weld (LGW)	Visual (VT-3) examination. Reinspection every 10 years following initial inspection.	25% of the total number of column assemblies (both visible and non-visible from above the lower core plate) using a VT-3 examination from above the lower core plate. The inspection coverage must be evenly distributed across the population of column assemblies (Notes 3 and 4). See Figure 4-23.
Core Barrel Assembly W6.1.Barrel-former bolts	All plants	Cracking (IASCC, Fatigue) Aging Management (IE, Void Swelling and ISR)	W6.Baffle-former bolts (also refer to MRP 2018-002)	Volumetric (UT) examination. Reinspection every 10 years following initial inspection.	100% of accessible barrel-former bolts (Minimum of 75% of the total population). Accessibility may be limited by presence of thermal shield or neutron pads. See Figure 4-21.
Lower Support Assembly W6.2.Lower support column bolts	All plants	Cracking (IASCC, Fatigue) Aging Management (IE and ISR)	W6.Baffle-former bolts	Volumetric (UT) examination. Reinspection every 10 years following initial inspection.	100% of accessible LSC bolts (Minimum of 75% of the total population) or as supported by plant-specific justification. See Figure 4-22

Notes to Table 4-6:

1. Examination acceptance criteria and expansion criteria for the Westinghouse components are in Table 5-3.
2. Examination coverage requires examination of either the ID or the OD of the weld.
3. The stated minimum coverage requirement is the minimum if no significant indications are found. However, the Examination Acceptance criteria in Section 5 require that additional coverage must be achieved in the same outage if significant flaws are found. This contingency should be considered for inspection planning purposes.
4. Justification that adequate distribution of the inspection coverage has been achieved can be based on geometric or layout arguments. Possible examples include, but are not limited to, inspection of all column assemblies in one quadrant of the lower core plate (based on the azimuthal symmetry of the plate) or inspecting every fourth column across the entire plate.
5. A minimum coverage of 75% of the weld length on the surface being examined shall be achieved; however, for welds with limited access (Note 6), a minimum examination coverage of 50% of the weld length on the surface being examined shall be achieved.
6. Accessibility to the MAW and LAW may be limited by the thermal shield or neutron panels – no disassembly to achieve higher weld length coverage is required.

4.4 Existing Programs Component Requirements

Component items in the Existing Programs tables are those PWR internals for which current aging and/or plant asset management activities are being generically implemented either by adherence to ASME Code Section XI or previous industry recommendations to address degradation during the original 40-year operating period. The continuation of these activities for the specific component items delineated in Tables 4-7, 4-8 and 4-9 is credited within these guidelines as adequate for continued aging management during the extended period of operation. See Section 4.2.7 regarding coordination of the Code and MRP inspections. The expectation of these guidelines is that these examinations will be conducted in accordance with the Code as well as the associated standards and practices that are normally invoked in such inspections.

For PWRs, ASME Code Section XI specifies examination of the vessel interior per Table IWB-2500-1 [2]. There are three sub-categories of Examination Category B-N in the table Titled B-N-1 "Interior of Reactor Vessel", B-N-2 Welded Core Support Structures and Interior Attachments to Reactor Vessels", and B-N-3 "Removable Core Support Structures". Only interior attachments within and outside the beltline (and not welded core support structures) are applicable to PWRs under B-N-2, B13.50 and B13.60 respectively. B-N-3 is only applicable to PWRs under B13.70. B13.70, Note 2 specifies that the core support structure be removed for examination and deferral of the examination to the end of interval is permissible for all but B-N-1 which is required each examination period.

Because the removal of the core support structure (i.e., Core Barrel/Core Support Assembly/Core Support Barrel) is a major evolution for plants it is commonly performed once per 10-year interval in conjunction with ASME Code Section XI volumetric weld examinations required for pressure retaining welds in reactor vessel and reactor vessel nozzles. Thus, ASME Code examinations associated with core support structure removal have typically been coordinated into a single plant outage designated the 10-year inservice inspection or "10-year ISI".

Component items associated with the removable core support structure are typically examined during the 10-year ISI under B-N-3 but additional vessel interior items (including non-removable components with core support function) are also rendered accessible due to the removal of the core support structure and are typically included in the examination scope. Regardless of whether a component item is designated B-N-1 or B-N-3, the requirements of B-N-1 and B-N-3 (visual VT-3) are considered sufficient to monitor for the aging effects addressed by these guidelines. Included in the Existing Programs are PWR internals that are typically examined during the "10-year ISI" whether as removable core support structures or internals. ASME Section XI, IWB-2500, Examination Category B-N [2] does not provide component-specific nomenclature associated with the reactor interior examination requirements. Accordingly, factors such as original design, licensing, code of construction and documentation (clerical) variability can result in significant differences in how an individual plant's current Category B-N-1/B-N-2/B-N-3 requirements are

fulfilled. Regardless of classification, however, the basic inspection requirements and effectiveness are considered equivalent. These guidelines therefore credit examination of specific components contained within the general Category B-N-1/B-N-2/B-N-3 classification for monitoring age-related degradation.

Table 4-8 lists the Existing examinations that are credited for aging management in the Combustion Engineering design plants. Many of these items may be considered core support structures that are typically examined during the 10-year inservice inspection per ASME Code Section XI Table IWB-2510 [2]. For these component items, the requirements of the Code (visual VT-3) are considered sufficient to monitor for the aging effects addressed by these guidelines.
Table 4-9 lists the Existing examinations that are credited for aging management in the Westinghouse-design plants. Many of these component items are considered core support structures that are typically examined during the 10-year inservice inspection per ASME Code Section XI Table IWB-2510 [2]. For these component items, the Code requirements for visual VT-3 are considered sufficient to monitor for the aging effects addressed by these guidelines.

Within the Westinghouse fleet there are three distinct upper support plate assembly designs (deep beam, top hat and inverted top hat). In each design the assembly is comprised of a distinct set of components. The original screening process identified SCC at structural welds and fatigue as potential degradation mechanisms. The location of potential degradation is dependent on the design, but generally occurs in the structural plates (flanges, rings, skirts, etc.) The top or outer surfaces of these structural plates should be readily available for visual inspection when the upper internals assembly is removed from the core barrel. In all three designs, the assembly is considered part of the core support structure and inspection of the accessible surfaces would be required by the ASME Code. Irradiation embrittlement is not a concern in the upper support assembly and the VT-3 examination should be sufficient for monitoring for cracking.

Also included in Existing Programs are those components for which existing guidance has been issued (e.g., from industry response fleet operating experience) to address degradation experienced during the original operating life of the PWR fleet. The continued implementation of this guidance has been determined to adequately manage the aging effects for these components.

A plant-specific Westinghouse-design item that is managed within the plant program is guidance for flux thimble tubes. The bottom mounted instrumentation flux thimble tubes (applicable to all plants) are inspected to ensure pressure boundary integrity, as initially prompted by Bulletin IEB 88-09 [45].

These examination requirements, as applied to the components designated in Tables 4-7, 4-8, and 4-9, have been determined to provide sufficient aging management for these components.

4.4.1 B&W Components

Table 4-7
B&W Plants Existing Programs Components

There are no Existing Programs components for B&W plants.

No existing generic industry programs were considered sufficiently specific for monitoring the aging effects addressed by these guidelines for B&W plants. Therefore, no components for B&W plants were placed into the Existing Programs group, and there is no Table 4-7.

4.4.2 CE Components

Table 4-8 describes the PWR internals in the Existing Programs for CE plants.

4.4.3 Westinghouse Components

Table 4-9 describes the PWR internals in the Existing Programs for Westinghouse plants.

4.5 No Additional Measures Components

It has been determined that no additional aging management is necessary for components in this group. Certain components designated as not requiring aging management may still be subject to ASME Code Section XI IWB-2500 [2] inservice inspection requirements. In no case does a determination of no inspection for aging management in this guideline relieve utilities of any ASME Code Section XI IWB-2500 inservice inspection requirement unless specific relief is granted as allowed by 10 CFR 50.55a [4].

A plant-specific Westinghouse-design item that has been managed by modification is the upper internals guide tube support pins (split pins). The degradation of split pins is not a safety issue, but it is an asset management concern. In Westinghouse-design plants, the originally installed Alloy X-750 split pins have typically been replaced with components with improved designs and less susceptible materials, such as Type 316 stainless steel. Type 316 stainless steel split pins are a Category A component, which is assigned to components for which the aging effects are below the screening criteria or for which aging degradation significance is minimal. Therefore, the Type 316 stainless steel split pins are considered a "no additional measures component" and do not require a plant-specific aging management program. For this reason, the Type 316 stainless steel split pins are not included in Table 4-9.

The plant owner should review their specific design, upgrade status, and asset management plans for guide tube split pins. If the originally installed Alloy X-750 split pins have not been replaced by components with a less susceptible material, a plant-specific evaluation shall be performed to determine an appropriate aging management program for the component and shall be included in the plant-specific RVI program for the period of extended operation, or until replacement is performed.

Table 4-8
CE Plants Existing Programs Components

Item	Applicability	Effect (Mechanism)	Reference	Examination Method	Examination Coverage
C13.Core Shroud Assembly Guide lugs	All plants	Loss of material (Wear) Aging Management (ISR)	ASME Code Section XI	Visual (VT-3) examination, general condition examination for detection of excessive or asymmetrical wear.	Accessible surfaces at specified frequency.
C14.Upper Internals Assembly Guide lug inserts and bolts	All plants	Loss of material (Wear) Aging Management (ISR)	ASME Code Section XI	Visual (VT-3) examination, general condition examination for detection of excessive or asymmetrical wear.	Accessible surfaces at specified frequency.
C15a.Lower Support Structure Fuel alignment pins	All plants with core shrouds assembled with full-height shroud plates	Cracking (SCC, IASCC, Fatigue) Aging Management (IE and ISR)	ASME Code Section XI	Visual (VT-3) examination to detect severed fuel alignment pins, missing locking tabs, or excessive wear on the fuel alignment pin nose or flange.	Accessible surfaces at specified frequency.
C15b.Lower Support Structure Fuel alignment pins	All plants with core shrouds assembled in two vertical sections	Loss of material (Wear) Aging Management (IE and ISR)	ASME Code Section XI	Visual (VT-3) examination.	Accessible surfaces at specified frequency.
C16.Core Barrel Assembly Upper flange	All plants	Loss of material (Wear)	ASME Code Section XI	Visual (VT-3) examination.	Area of the upper flange potentially susceptible to wear.
C17.Alignment and Interfacing Components Core Stabilizing Lugs and Shims	All plants (TB-14-5)	Cracking (SCC) of bolting	ASME Code Section XI as supplemented by TB-14-5 (Note 1)	Visual (VT-3) examination.	Accessible surfaces at specified frequency

Note to Table 4-8:

1. The core stabilizing lugs are attached to the reactor vessel and the shims are bolted to the lugs. The ASME Code examination of accessible surfaces is considered to include all details of the shim configuration, including the bolting and bolt lock pins. The bolting is fabricated from nickel-based materials and is susceptible to stress corrosion cracking (SCC). Although failure of the bolting does not itself cause loss of support function, asset impairment or issues with core barrel removal are a subsequent possibility. Westinghouse technical bulletin TB 14-5 dated 8/25/2014 provides additional information regarding possible visual indications that shim bolting failure may have occurred. This information should be reviewed to ensure a heightened awareness of the examiners is applied to this Code inspection.

Table 4-9
Westinghouse Plants Existing Programs Components

Item	Applicability	Effect (Mechanism)	Reference	Examination Method	Examination Coverage
W10.Core Barrel Assembly Core barrel flange	All plants	Loss of material (Wear)	ASME Code Section XI	Visual (VT-3) exam to determine general condition for excessive wear.	All accessible surfaces at specified frequency.
W11.Upper Internals Assembly Upper support ring or skirt	All plants	Cracking (SCC, Fatigue)	ASME Code Section XI	Visual (VT-3) examination.	All accessible surfaces at specified frequency.
W12a.Lower Internals Assembly Lower core plate XL lower core plate (Note 1)	All plants	Cracking (IASCC, Fatigue) Aging Management (IE)	ASME Code Section XI as supplemented by TB-16-4	Visual (VT-3) exam of the lower core plates to detect evidence of distortion and/or loss of bolt integrity.	All accessible surfaces at specified frequency.
W12b.Lower Internals Assembly Lower core plate XL lower core plate (Note 1)	All plants	Loss of material (Wear)	ASME Code Section XI as supplemented by TB-16-4	Visual (VT-3) examination.	All accessible surfaces at specified frequency.
W13.Bottom Mounted Instrumentation System Flux thimble tubes	All plants	Loss of material (Wear)	IEB 88-09	Surface (ET) examination.	Eddy current surface examination as defined in plant response to IEB 88-09.
W14.Alignment and Interfacing Components Clevis bearing Stellite wear surface Clevis insert bolts (Note 2)	All plants (TB-14-5)	Loss of material (wear) Cracking (SCC)	ASME Code Section XI as supplemented by TB-14-5 (Note 2)	Visual (VT-3) examination.	All accessible surfaces at specified frequency.
W15.Alignment and Interfacing Components Upper core plate alignment pins	All plants	Loss of material (Wear)	ASME Code Section XI as supplemented by TB-16-4	Visual (VT-3) examination.	All accessible surfaces at specified frequency.

Notes to Table 4-9:

1. XL = "Extra Long" referring to Westinghouse plants with 14-foot cores.
2. The clevis inserts are attached to integrally welded reactor vessel lugs and the inserts are bolted to the lugs. The ASME Code examination of accessible surfaces is considered to include all details of the clevis configuration, including the bolting and locking devices. The bolting is fabricated from nickel-based materials and is susceptible to stress corrosion cracking (SCC). Although failure of the bolting does not itself cause loss of support function, asset impairment or issues with core barrel removal are a subsequent possibility. Westinghouse technical bulletin TB 14-5 dated 8/25/2014 provides additional information regarding possible visual indications that clevis bolting failure may have occurred. This information should be reviewed to ensure a heightened awareness of the examiners is applied to this Code inspection.

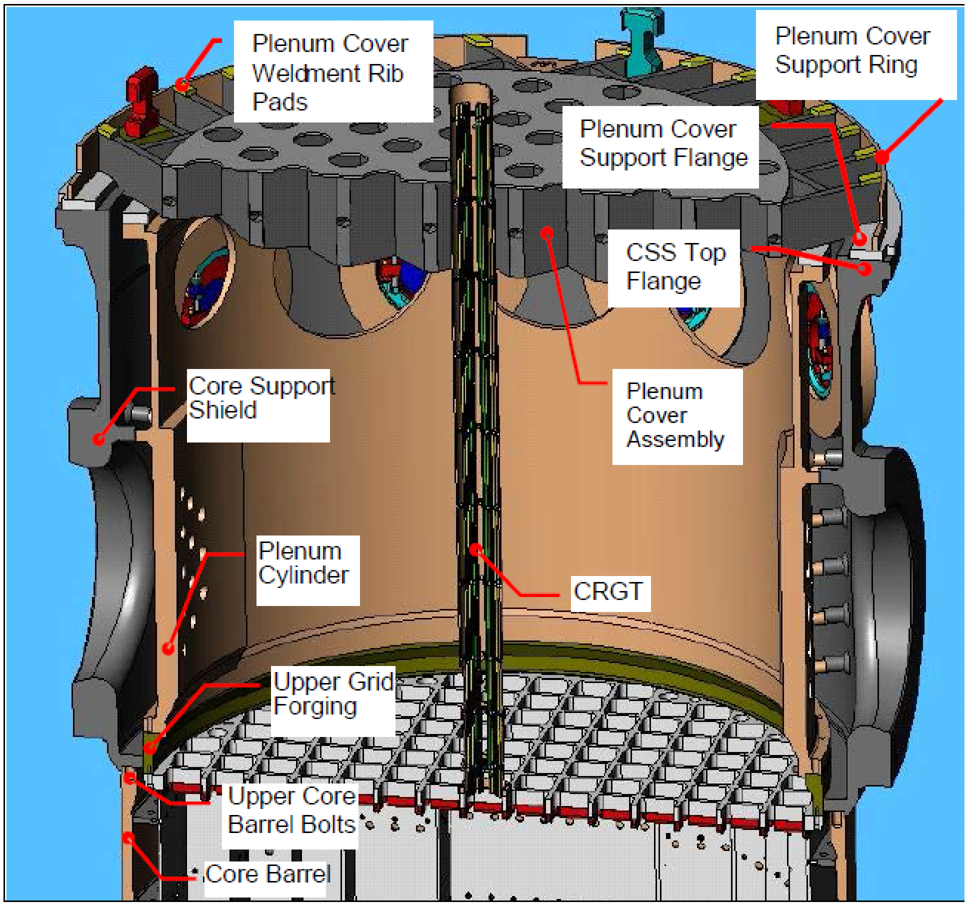

Figure 4-1
Typical Upper Internals Arrangement for B&W-Designed PWRs

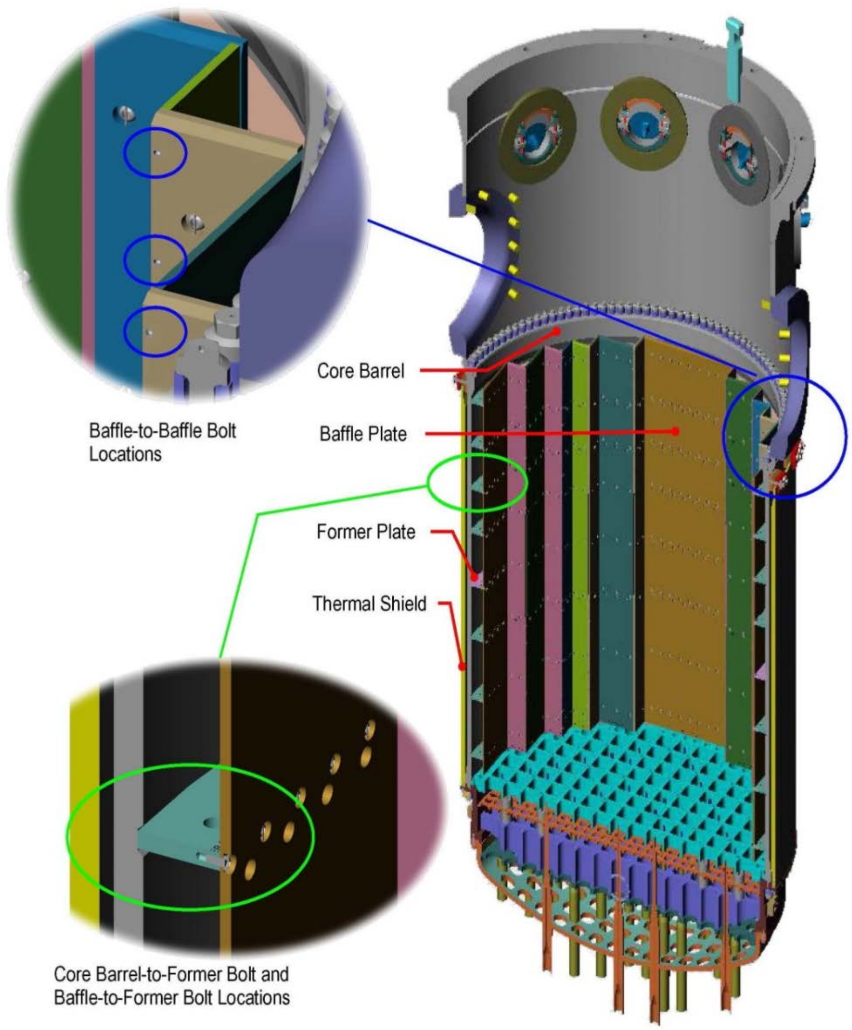

Figure 4-2
Typical Internals Core Barrel Assembly for B&W-Designed PWRs

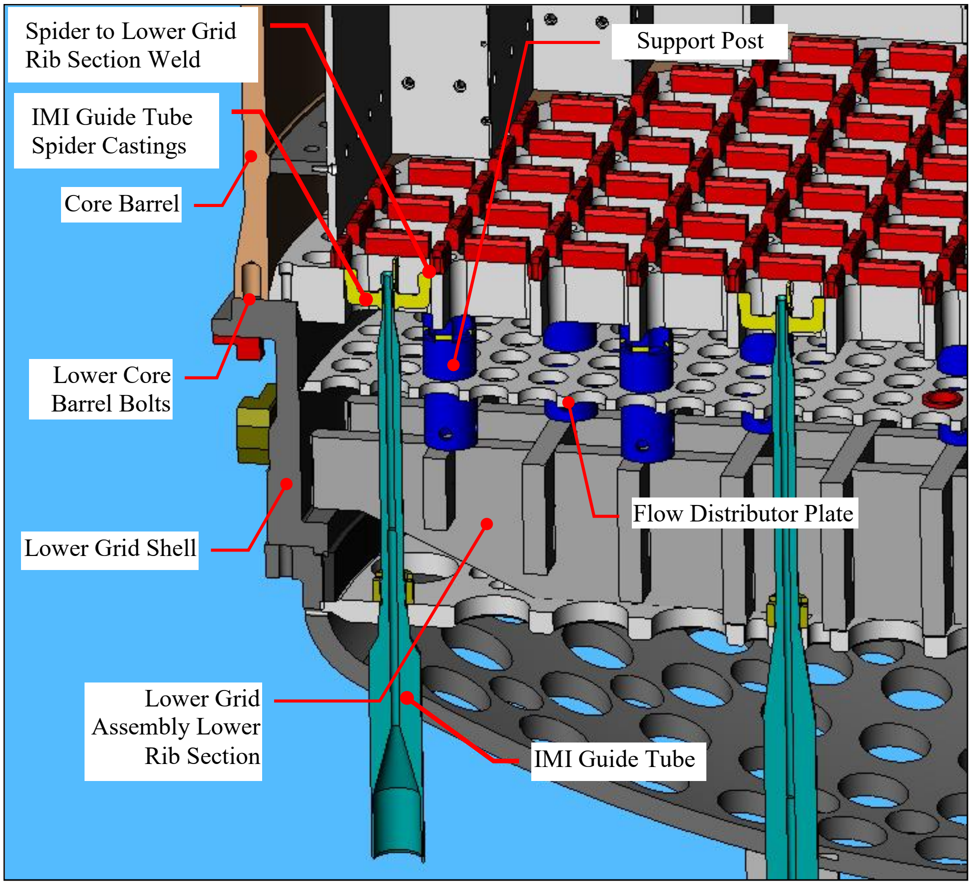

Figure 4-3
Typical Lower Internals Arrangement for B&W-Designed PWRs

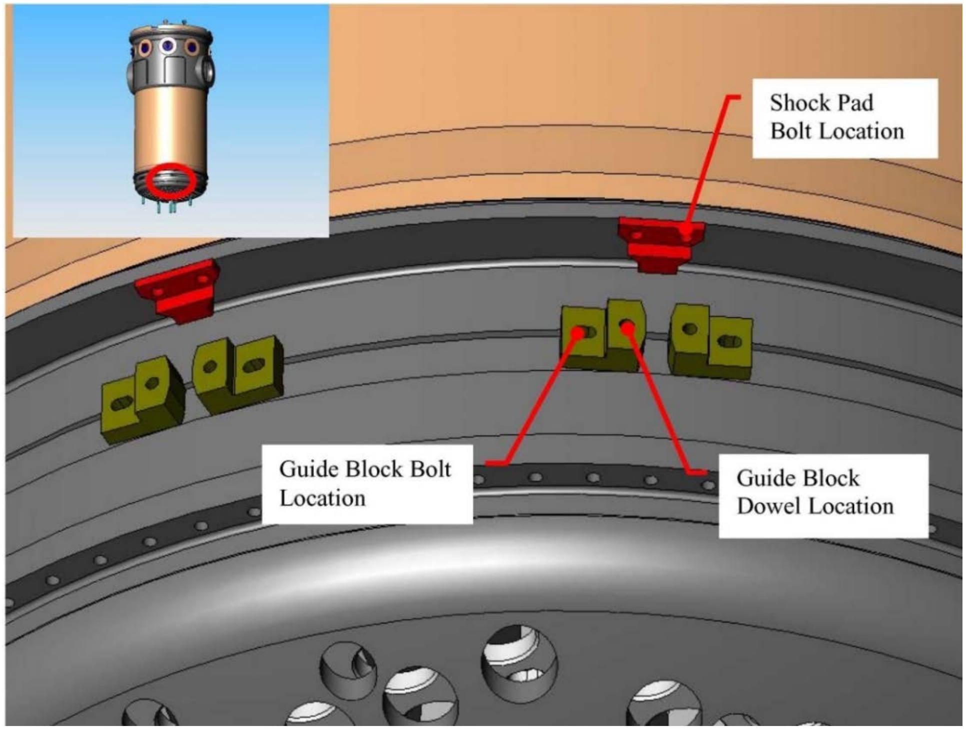

Figure 4-4
Typical Guide Block and Shock Pad Locations for B&W-Designed PWRs

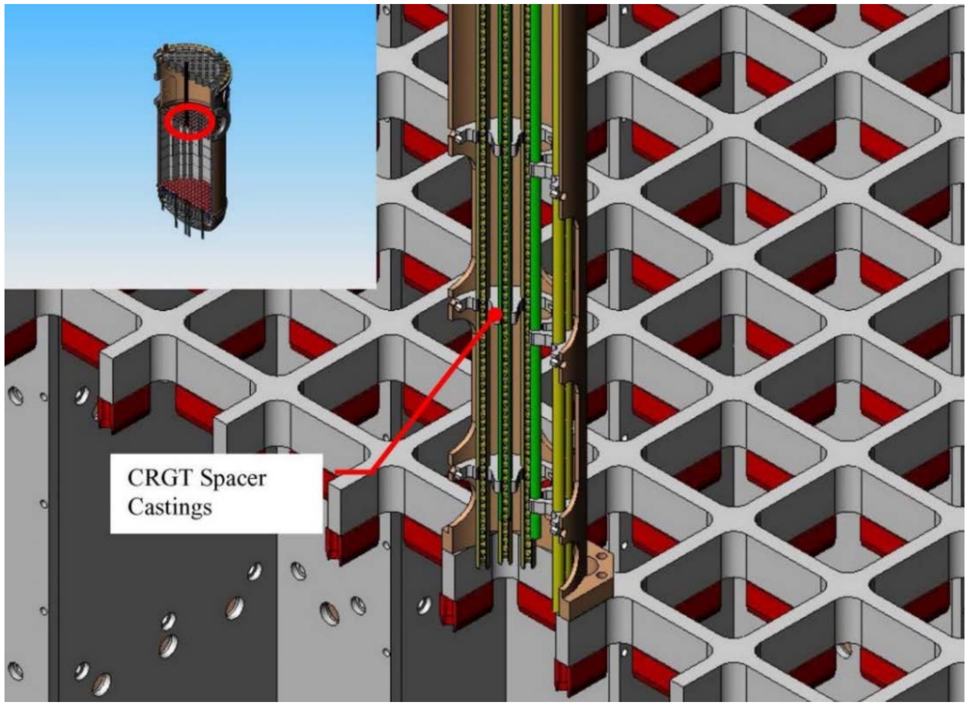

Figure 4-5
Typical Control Rod Guide Tube (CRGT) for B&W-Designed PWRs (1 of 69 CRGTs Shown)

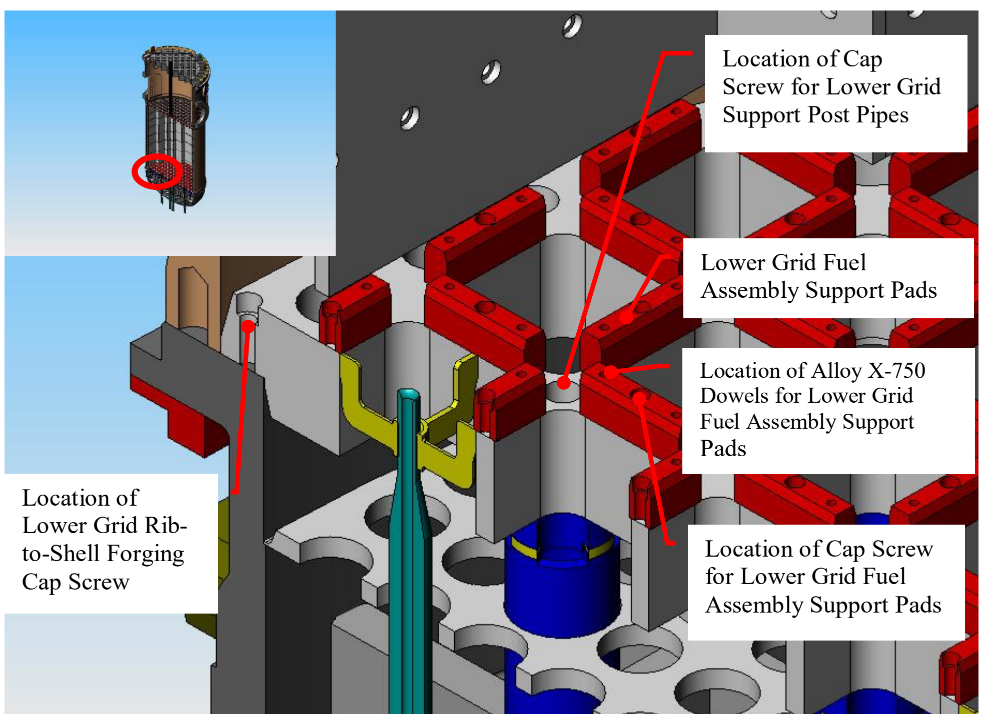

Figure 4-6
Typical Lower Grid Assembly and Fuel Assembly Support Pads for B&W-Designed PWRs

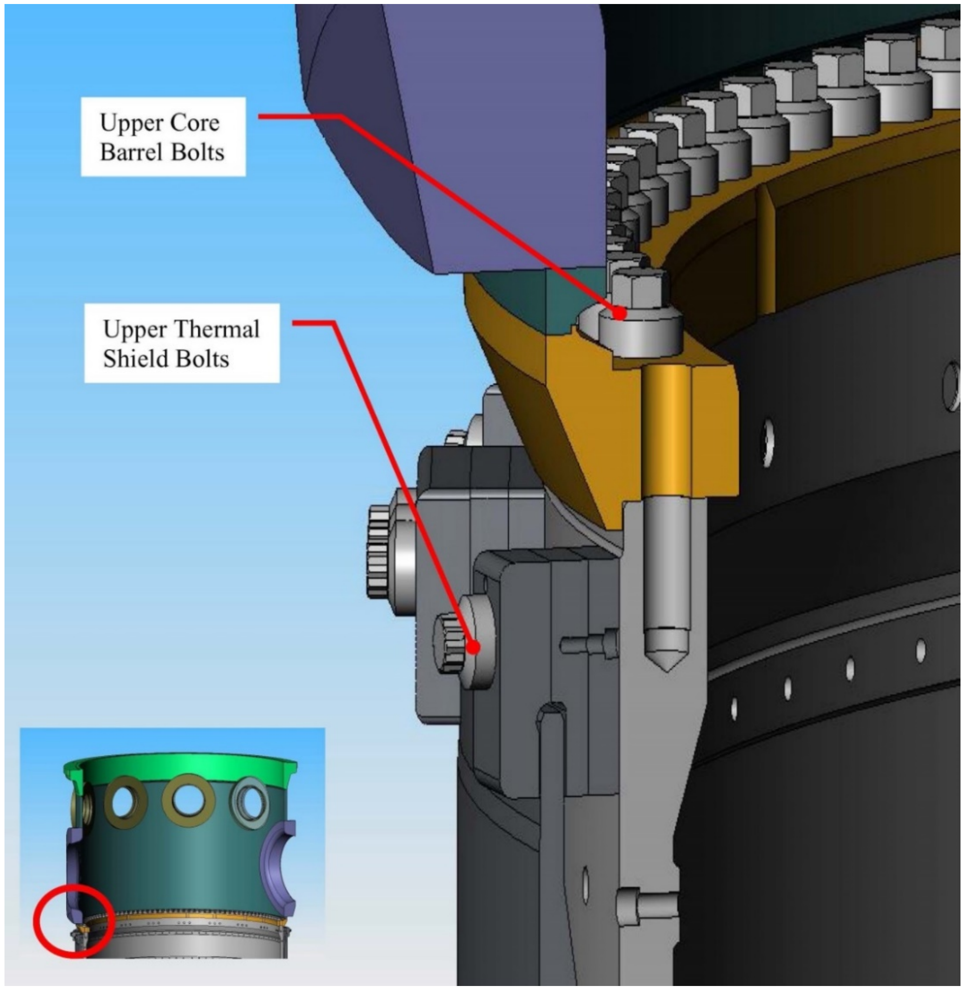

Figure 4-7
Typical Upper Thermal Shield Bolts and Upper Core Barrel Bolts for B&W-Designed PWRs

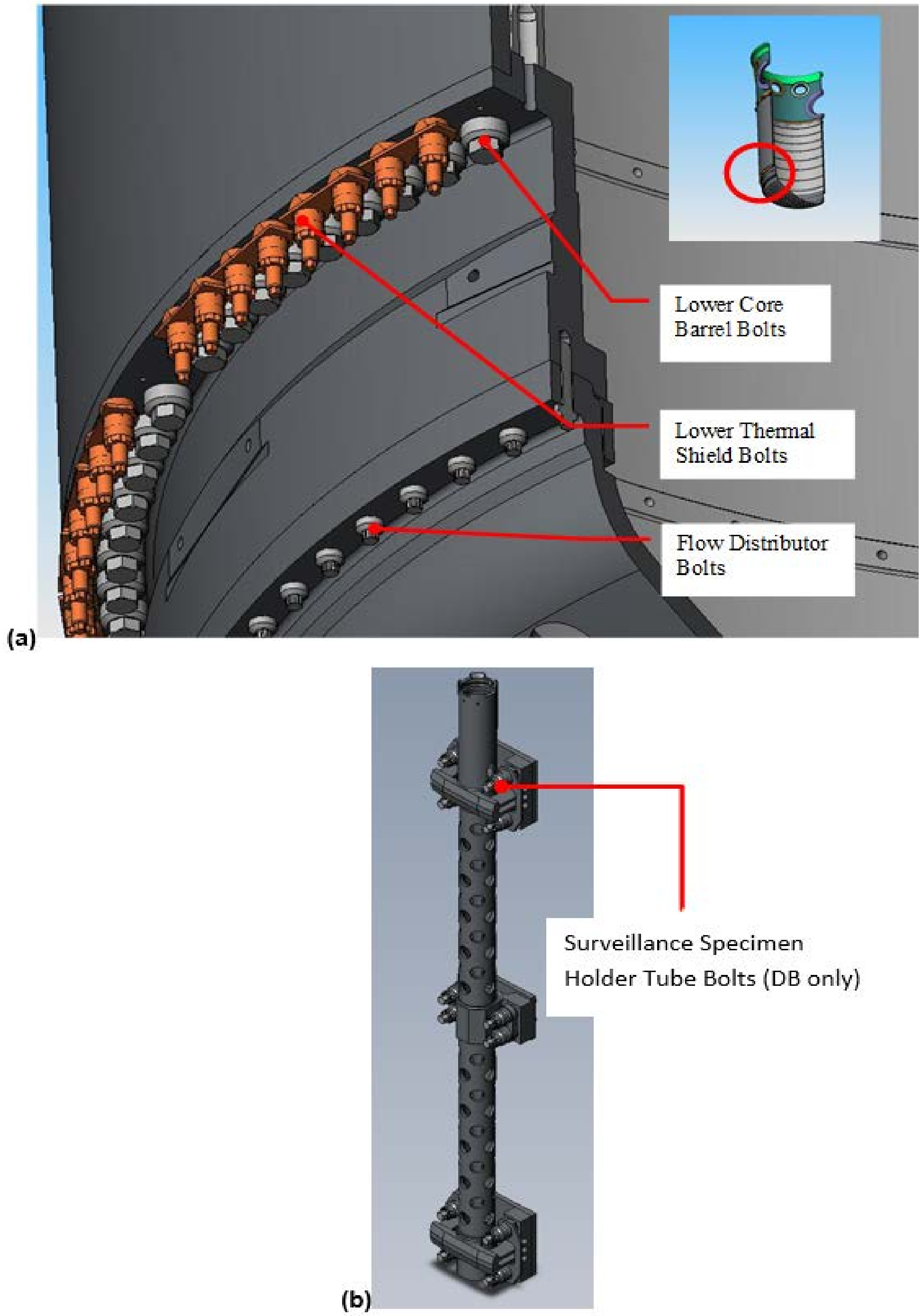

Figure 4-8
Typical (a) Lower Thermal Shield Bolts, Lower Core Barrel Bolts, and Flow Distributor Bolts, (b) Surveillance Specimen Holder Tube Bolts for B&W-Designed PWRs

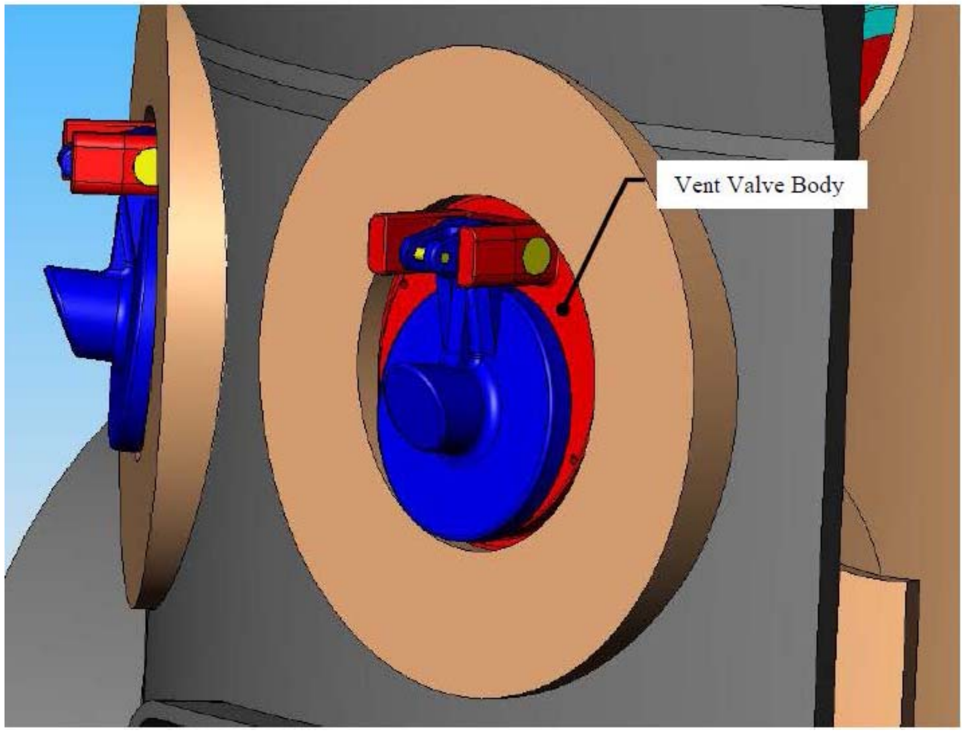

Figure 4-9
Typical Vent Valve – Outside View – for B&W-Designed PWRs

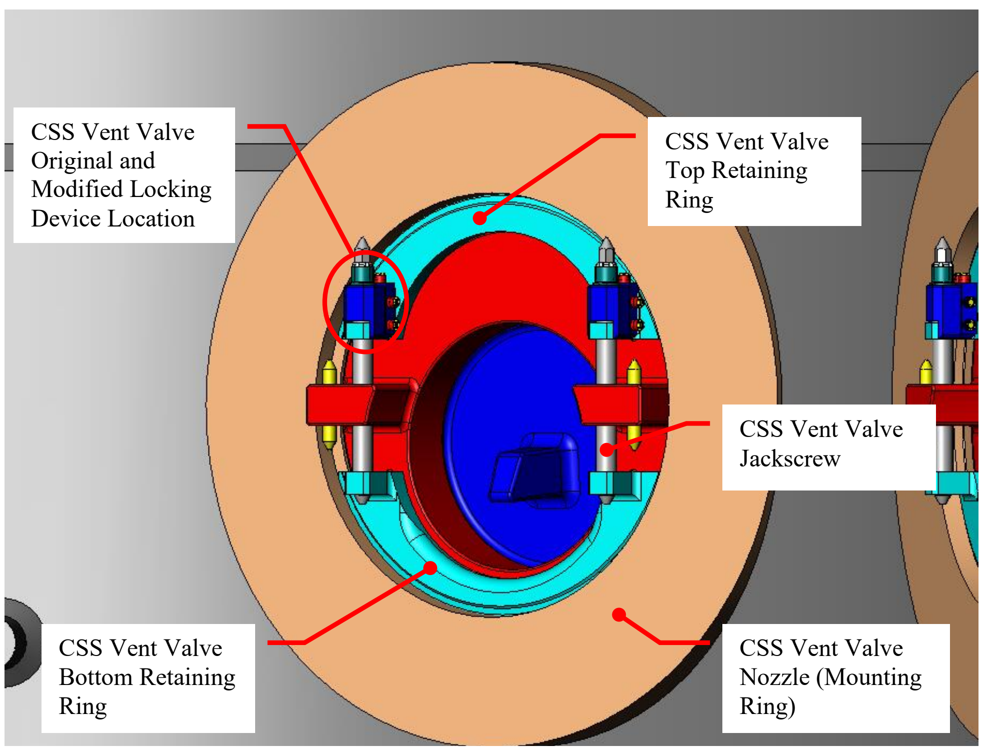

Figure 4-10
Typical Vent Valve – Inside View – for B&W-Designed PWRs

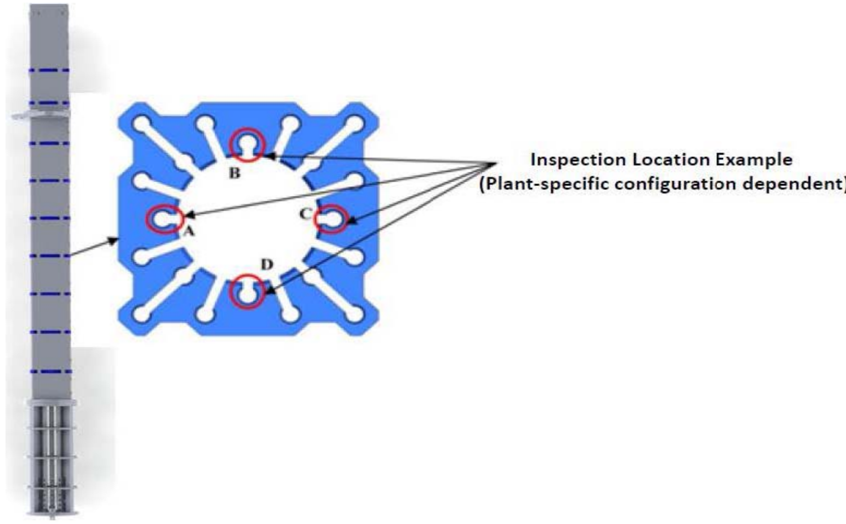

*Figure 4-11**
Typical Westinghouse-Design Control Rod Guide Tube Assembly Guide Plates (Cards)

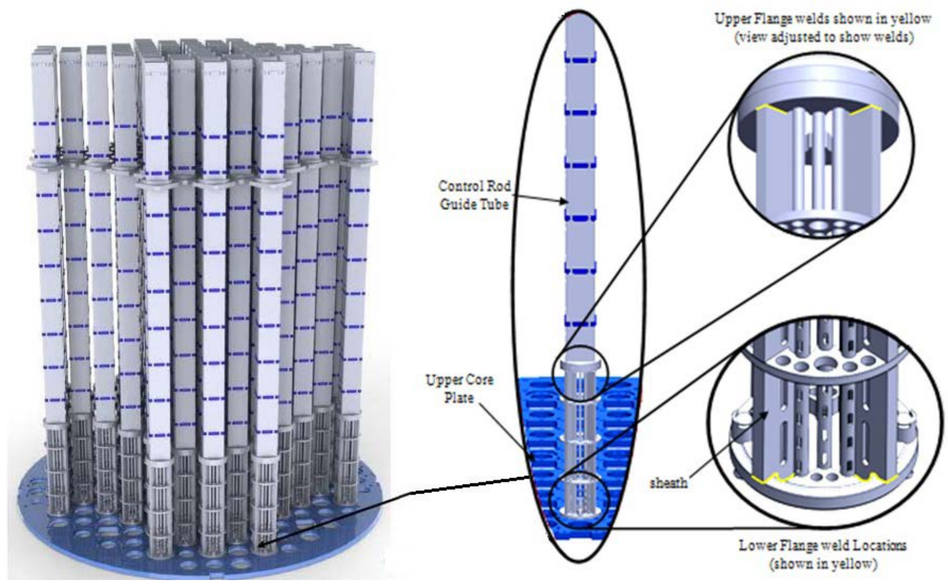

*Figure 4-12**
Typical Westinghouse-Design Control Rod Guide Tube Assembly Lower Flange Welds

* Note: Figure intended to represent the component features and relative position to other portions of the structure and system. Figure does not contain or transmit design information and is not meant to be used for performing engineering assessments, evaluations, or examinations.

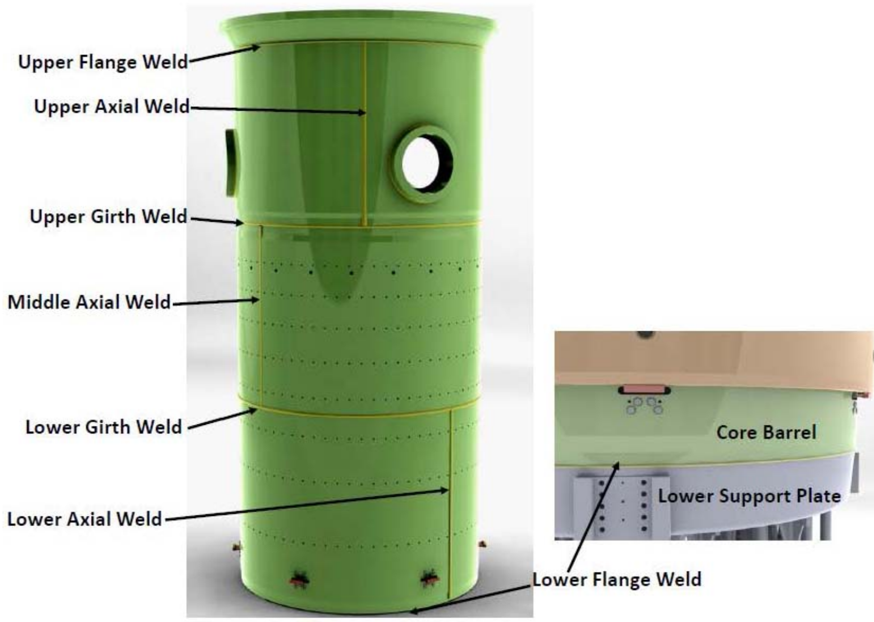

*Figure 4-13**
Typical Westinghouse-Design Core Barrel Assembly Core Barrel Welds

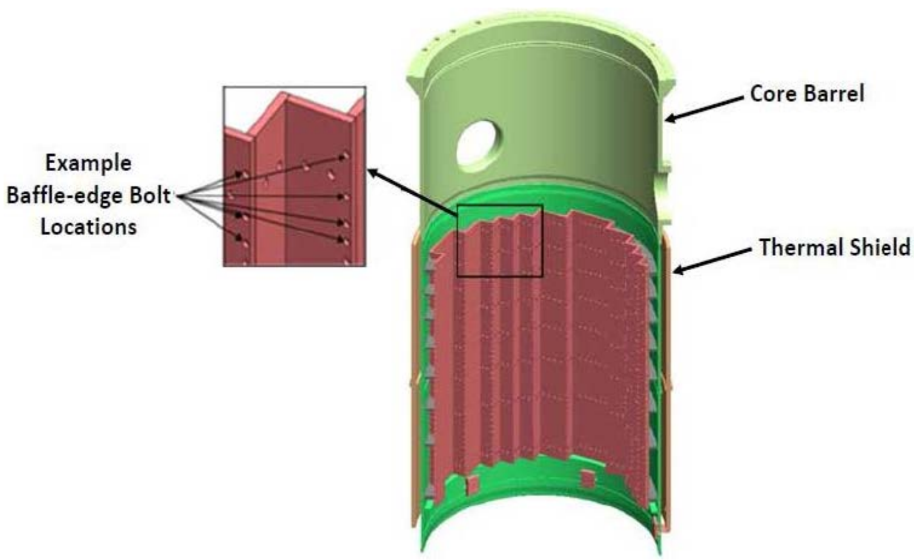

*Figure 4-14**
Typical Westinghouse-Design Baffle-Former Assembly Baffle-Edge Bolts

* Note: Figure intended to represent the component features and relative position to other portions of the structure and system. Figure does not contain or transmit design information and is not meant to be used for performing engineering assessments, evaluations, or examinations.

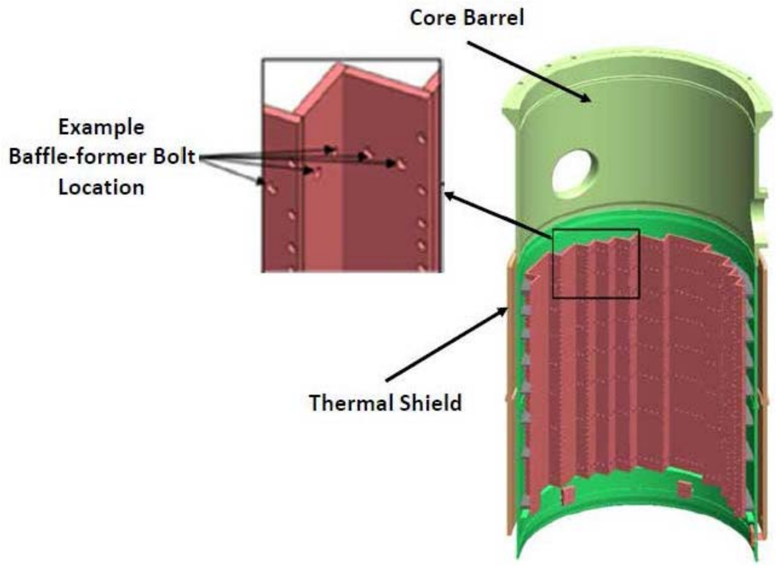

*Figure 4-15**
Typical Westinghouse-Design Baffle-Former Assembly Baffle-Former Bolts

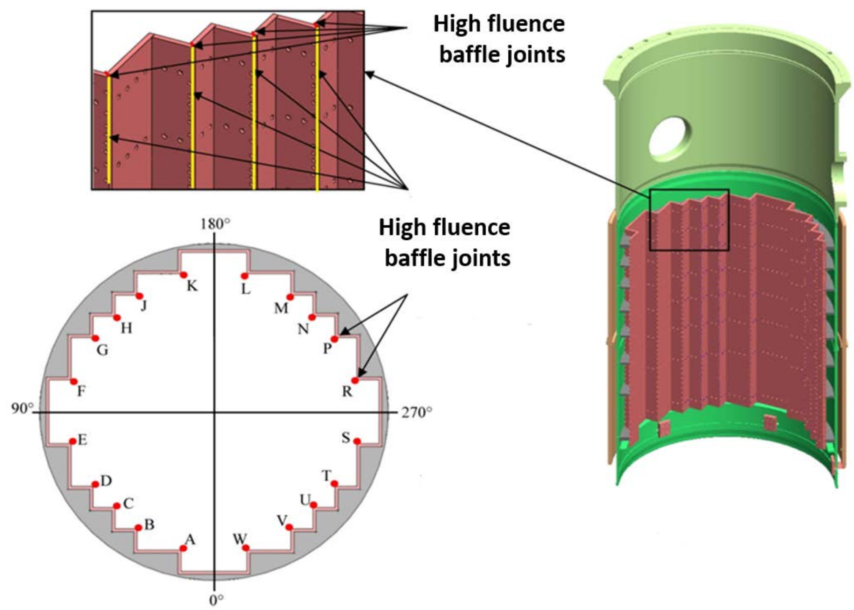

*Figure 4-16**
Typical Westinghouse-Design Baffle-Former Assembly

* Note: Figure intended to represent the component features and relative position to other portions of the structure and system. Figure does not contain or transmit design information and is not meant to be used for performing engineering assessments, evaluations, or examinations.

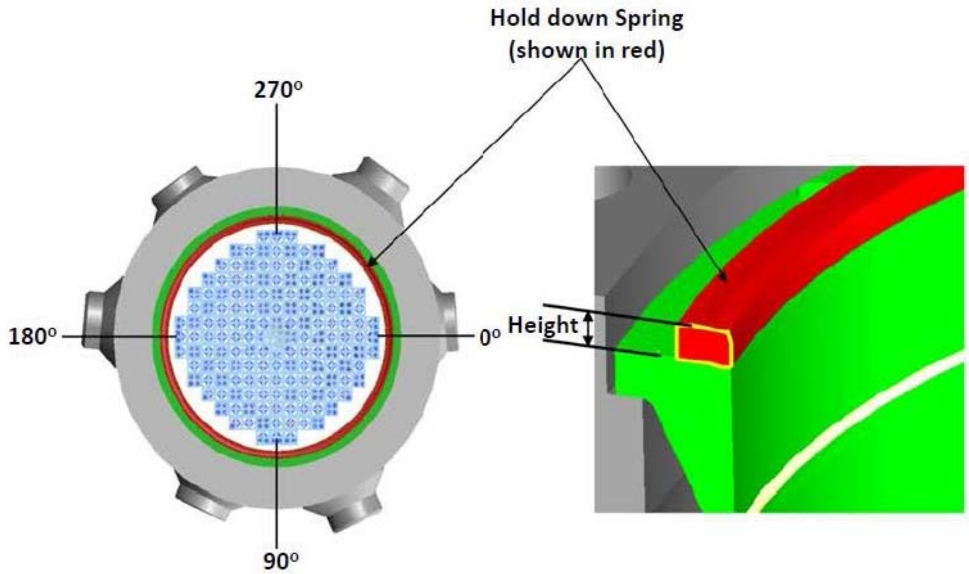

Figure 4-17
Typical Westinghouse-Design Alignment and Interfacing Components Internals Hold Down Spring

Note: Figure intended to represent the component features and relative position to other portions of the structure and system. Figure does not contain or transmit design information and is not meant to be used for performing engineering assessments, evaluations, or examinations.

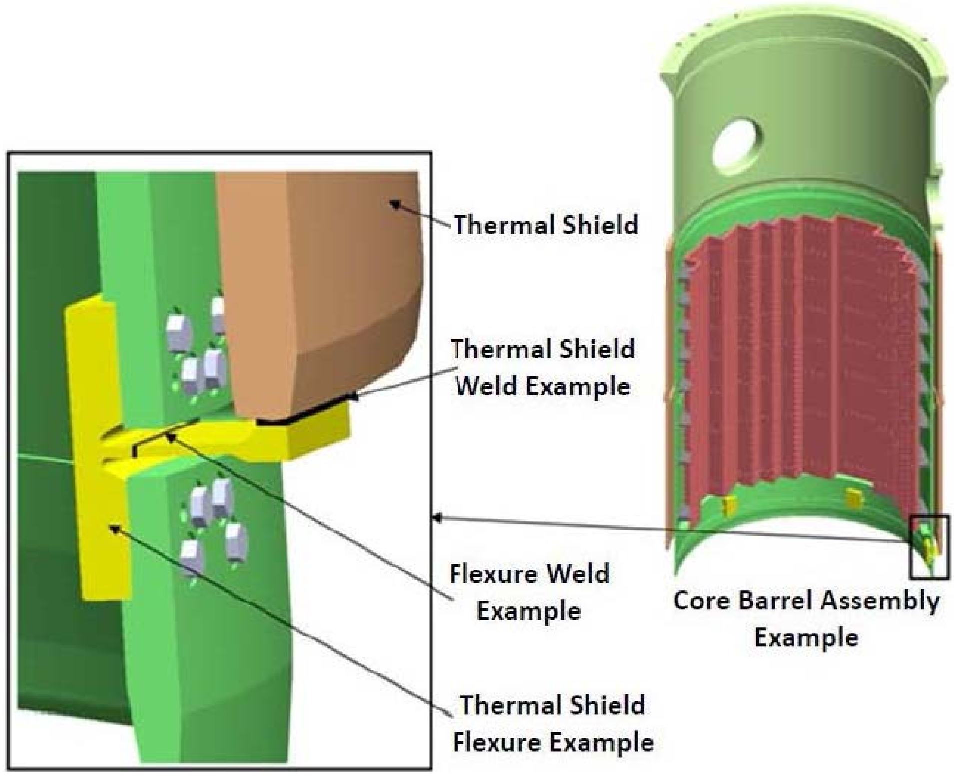

Figure 4-18
Typical Westinghouse-Design Thermal Shield Assembly Thermal Shield Flexures

Note: Figure intended to represent the component features and relative position to other portions of the structure and system. Figure does not contain or transmit design information and is not meant to be used for performing engineering assessments, evaluations, or examinations.

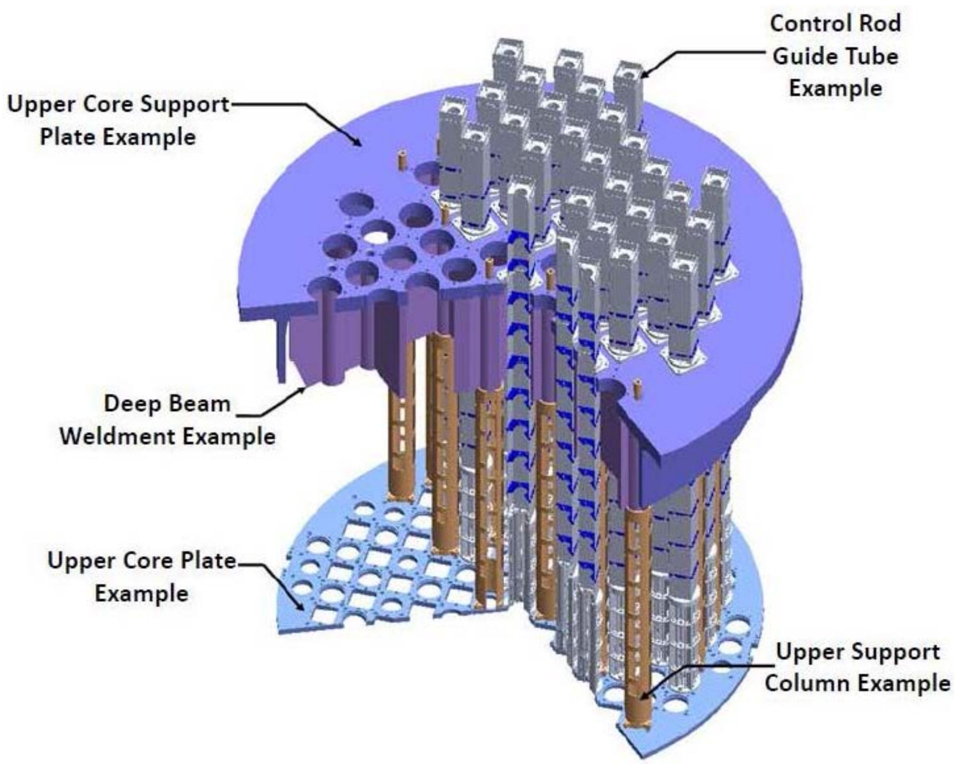

Figure 4-19
Typical Westinghouse-Design Upper Internals Assembly Upper Core Plate

Note: Figure intended to represent the component features and relative position to other portions of the structure and system. Figure does not contain or transmit design information and is not meant to be used for performing engineering assessments, evaluations, or examinations.

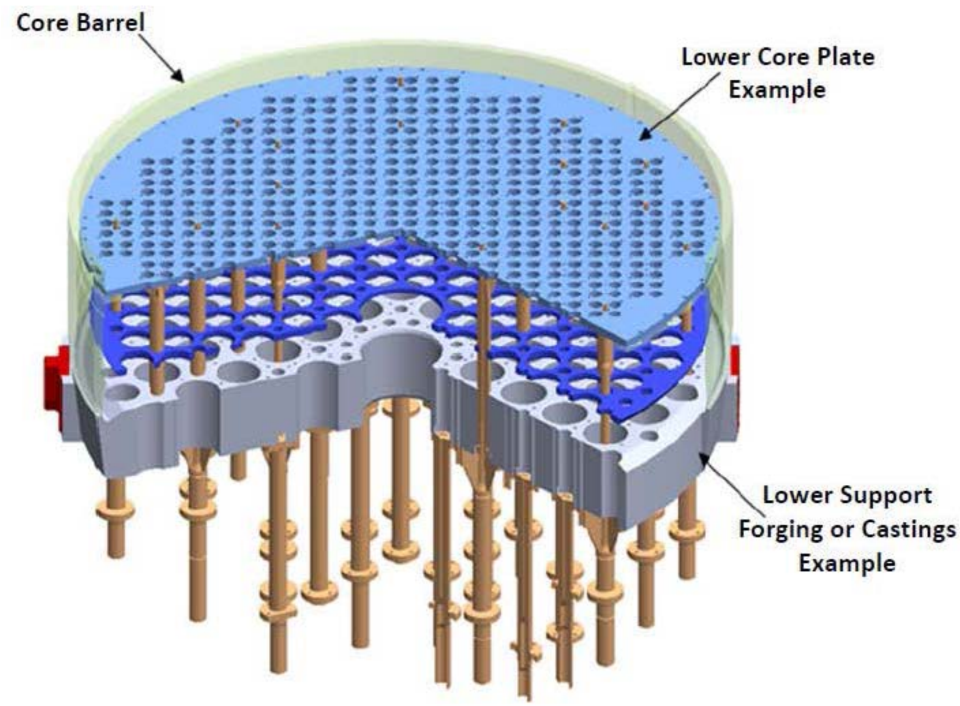

Figure 4-20
Typical Westinghouse-Design Lower Internals Assembly Lower Support Forging or Castings

Note: Figure intended to represent the component features and relative position to other portions of the structure and system. Figure does not contain or transmit design information and is not meant to be used for performing engineering assessments, evaluations, or examinations.

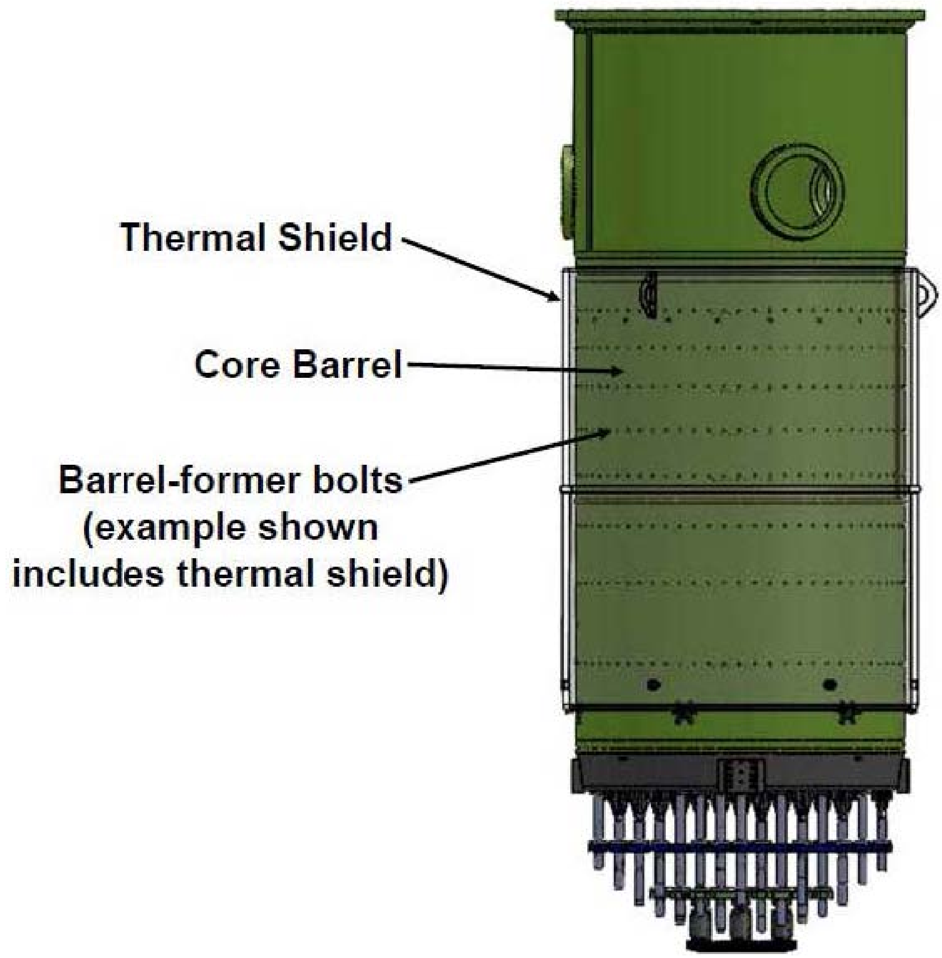

Figure 4-21
Typical Westinghouse-Design Core Barrel Assembly Barrel-former Bolts

Note: Figure intended to represent the component features and relative position to other portions of the structure and system. Figure does not contain or transmit design information and is not meant to be used for performing engineering assessments, evaluations, or examinations.

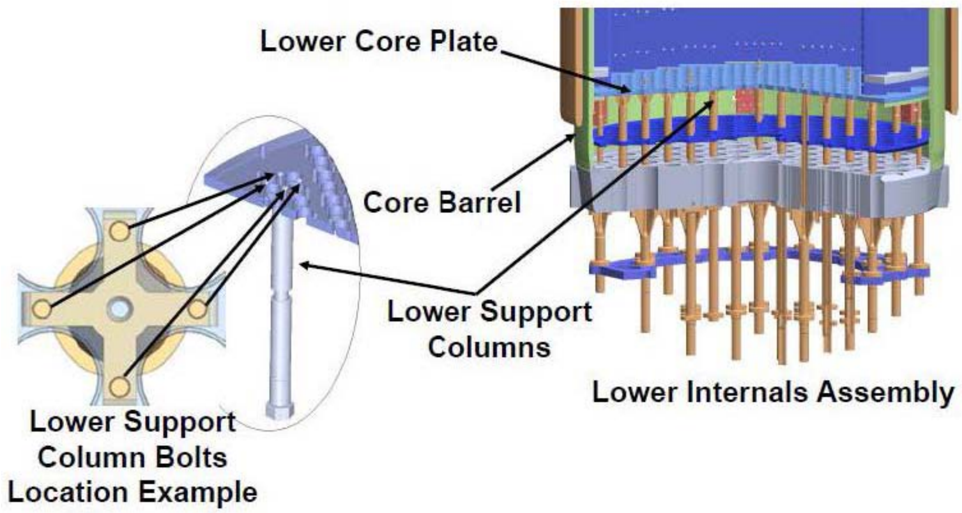

*Figure 4-22**
Typical Westinghouse-Design Lower Support Assembly Lower Support Column Bolts

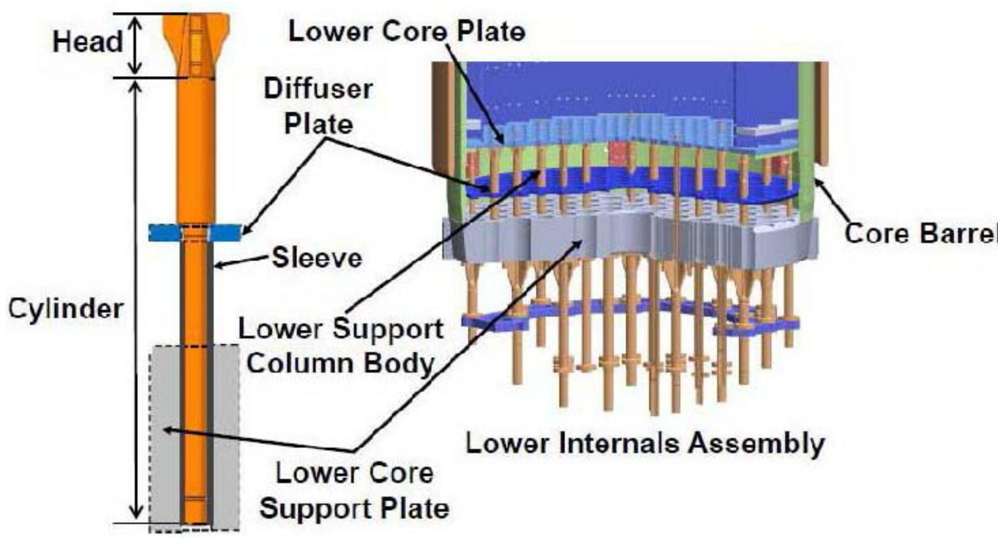

*Figure 4-23**
Typical Westinghouse-Design Lower Support Assembly Lower Support Column Bodies (Cast and Non-Cast)

* Note: Figure intended to represent the component features and relative position to other portions of the structure and system. Figure does not contain or transmit design information and is not meant to be used for performing engineering assessments, evaluations, or examinations.

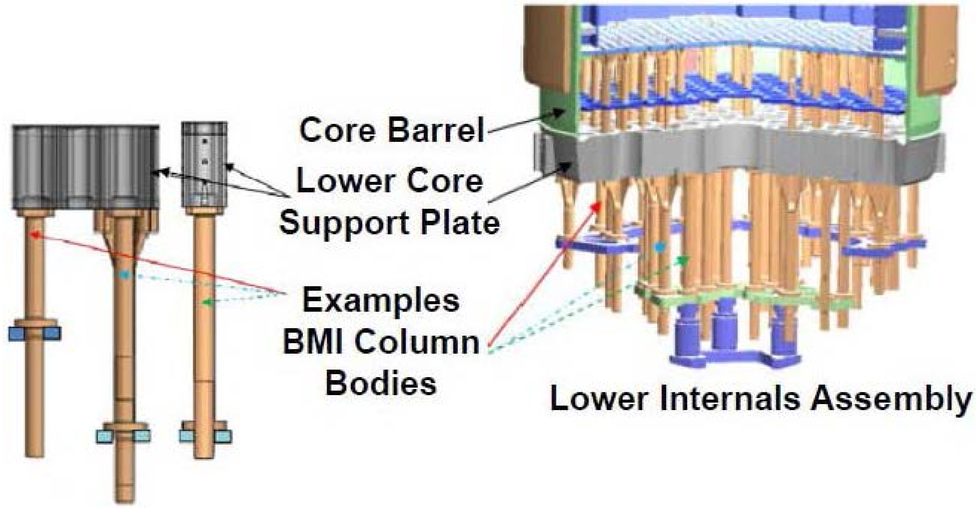

Figure 4-24
Typical Westinghouse-Design Lower Support Assembly Bottom-mounted Instrumentation (BMI) Column Bodies

Note: Figure intended to represent the component features and relative position to other portions of the structure and system. Figure does not contain or transmit design information and is not meant to be used for performing engineering assessments, evaluations, or examinations.

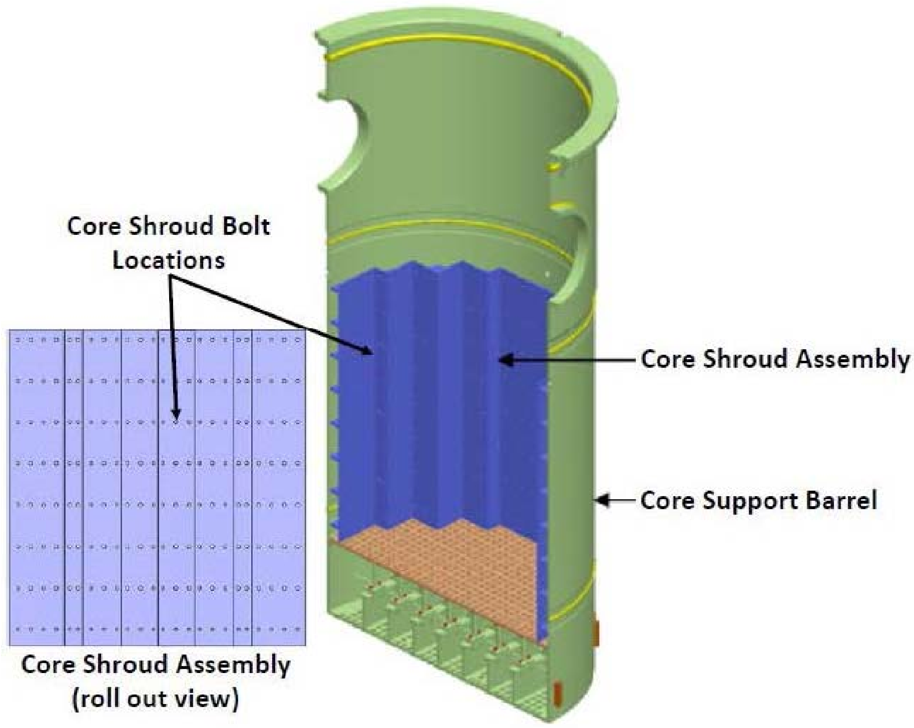

Figure 4-25
Typical CE-Design Core Shroud Assembly (Bolted) Core Shroud Bolts

Note: Figure intended to represent the component features and relative position to other portions of the structure and system. Figure does not contain or transmit design information and is not meant to be used for performing engineering assessments, evaluations, or examinations.

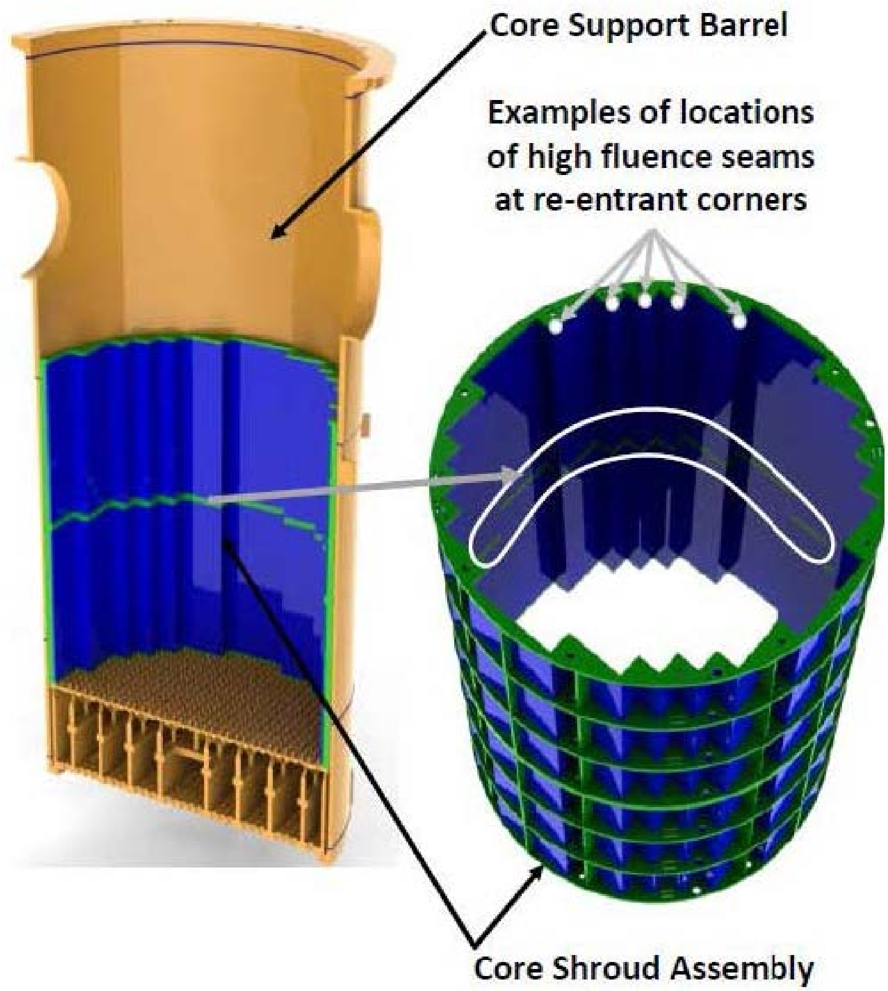

Figure 4-26
Typical CE-Design Core Shroud Assembly (Welded) Assembly and Core Shroud Plate-Former Plate Weld

Note: Figure intended to represent the component features and relative position to other portions of the structure and system. Figure does not contain or transmit design information and is not meant to be used for performing engineering assessments, evaluations, or examinations.

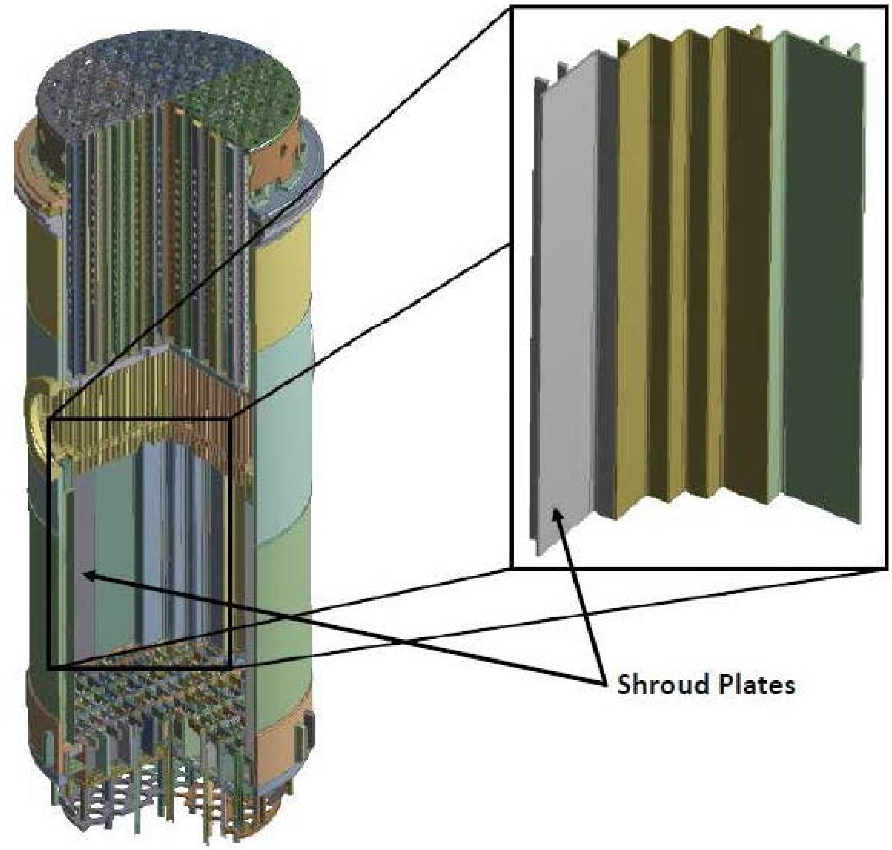

Figure 4-27
Typical CE-Design Core Shroud Assembly (Welded) Shroud Plates

Note: Figure intended to represent the component features and relative position to other portions of the structure and system. Figure does not contain or transmit design information and is not meant to be used for performing engineering assessments, evaluations, or examinations.

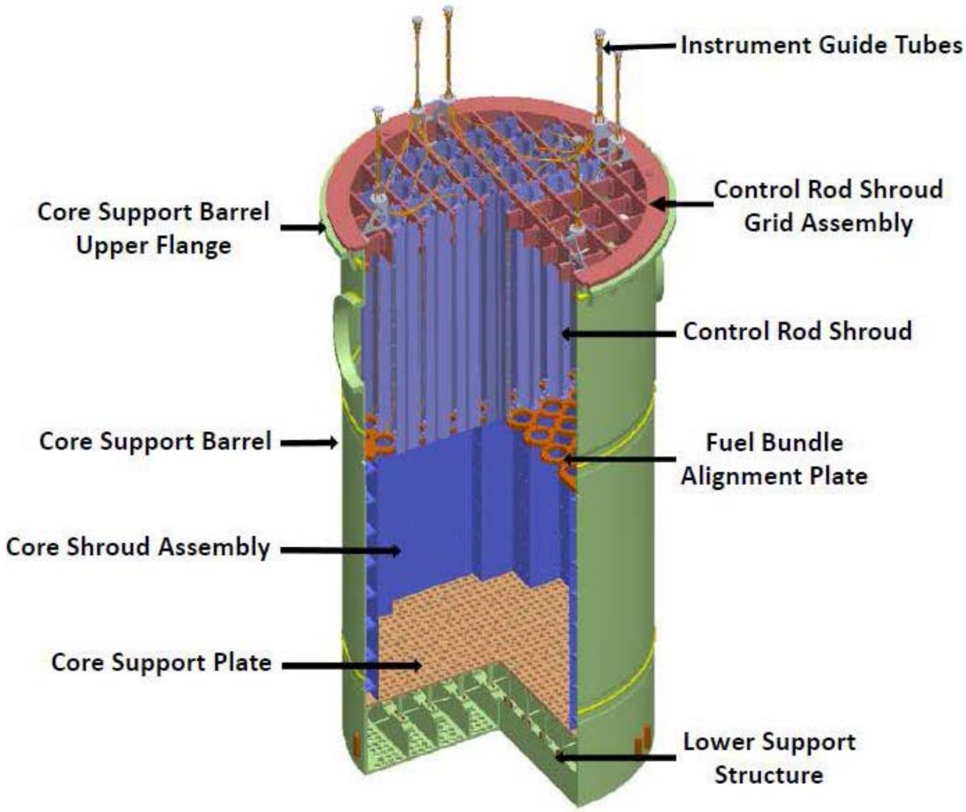

Figure 4-28
Typical CE-Design Core Shroud Assembly (Bolted) Assembly

Note: Figure intended to represent the component features and relative position to other portions of the structure and system. Figure does not contain or transmit design information and is not meant to be used for performing engineering assessments, evaluations, or examinations.

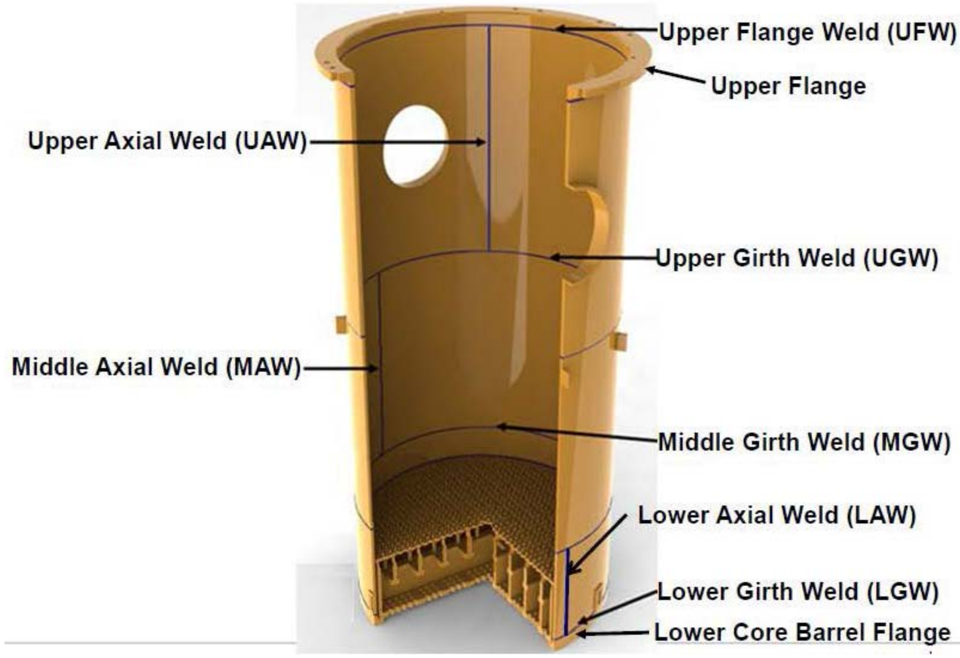

Figure 4-29
Typical CE-Design Core Support Barrel Assembly - Welds, Upper Cylinder (Including Welds), Upper Core Barrel Flange, Core Barrel Assembly Axial Welds, Lower Core Barrel Flange

Note: Figure intended to represent the component features and relative position to other portions of the structure and system. Figure does not contain or transmit design information and is not meant to be used for performing engineering assessments, evaluations, or examinations.

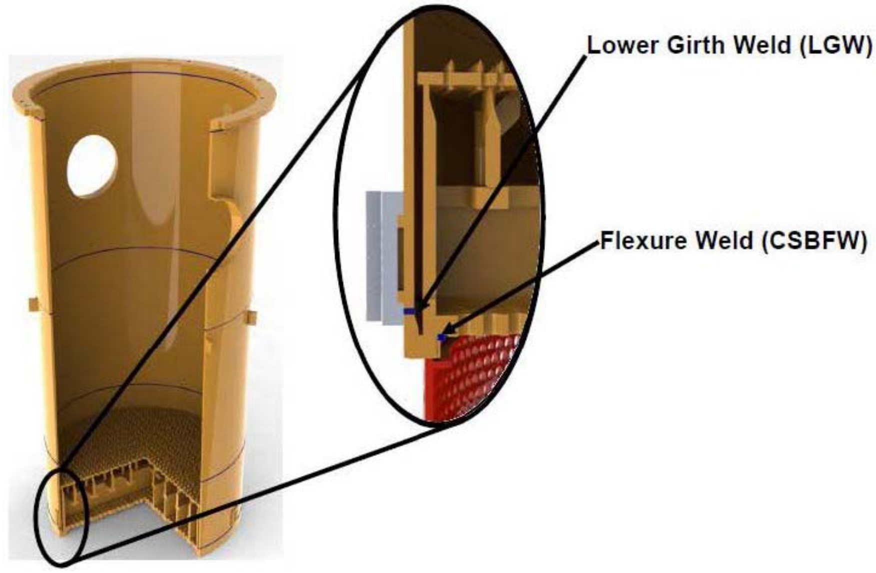

Figure 4-30
Typical CE-Design Core Support Barrel Assembly CSB Flexure Weld (CSBFW)

Note: Figure intended to represent the component features and relative position to other portions of the structure and system. Figure does not contain or transmit design information and is not meant to be used for performing engineering assessments, evaluations, or examinations.

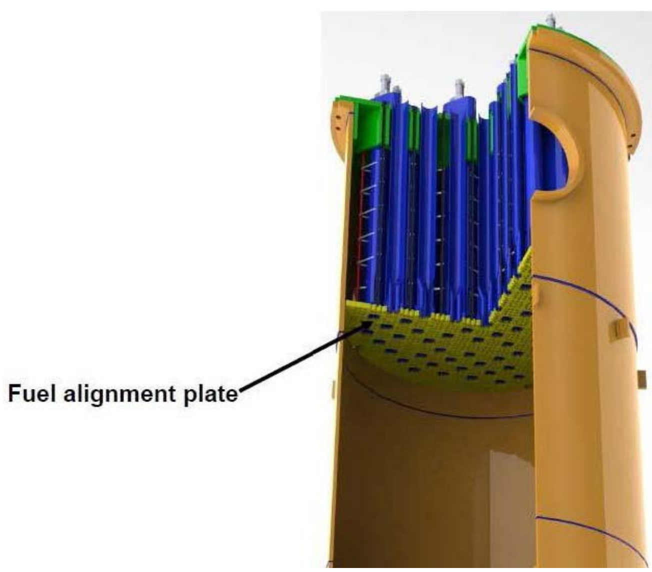

Figure 4-31
Typical CE-Design Upper Internals Assembly Fuel Alignment Plate

Note: Figure intended to represent the component features and relative position to other portions of the structure and system. Figure does not contain or transmit design information and is not meant to be used for performing engineering assessments, evaluations, or examinations.

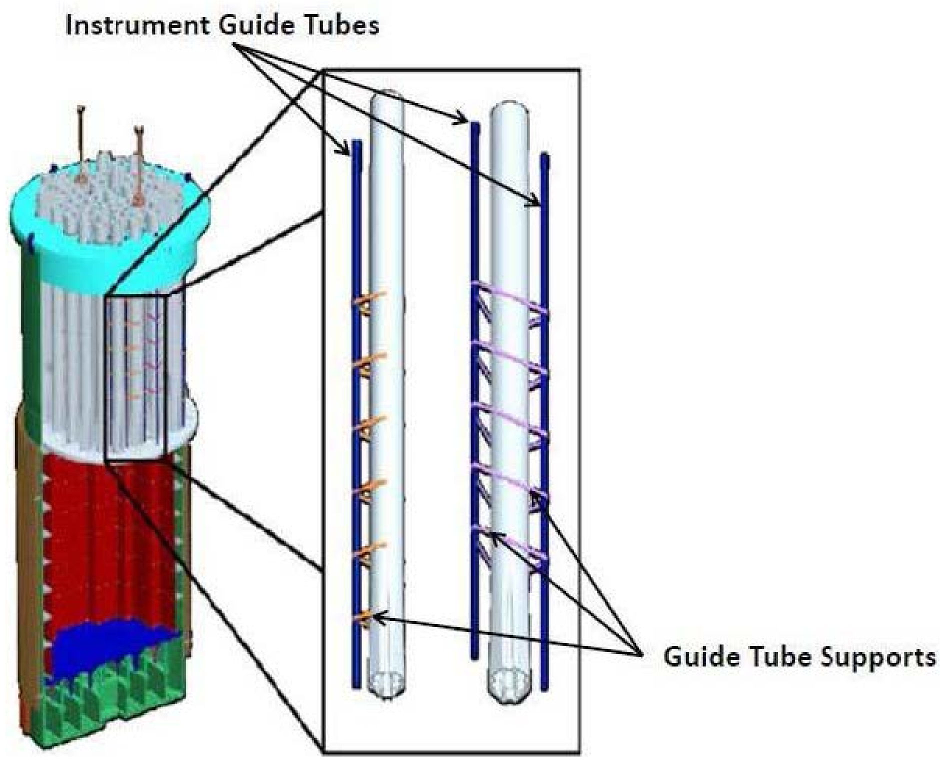

Figure 4-32
Typical CE-Design Control Element Assembly Instrument Guide Tubes

Note: Figure intended to represent the component features and relative position to other portions of the structure and system. Figure does not contain or transmit design information and is not meant to be used for performing engineering assessments, evaluations, or examinations.

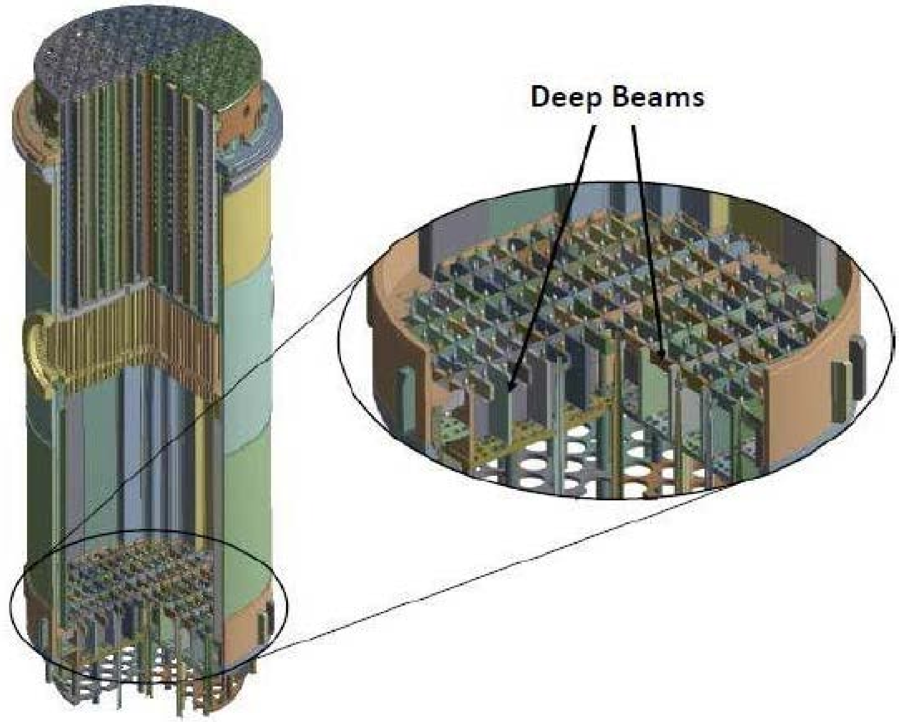

Figure 4-33
Typical CE-Design Lower Support Structure Deep Beams

Note: Figure intended to represent the component features and relative position to other portions of the structure and system. Figure does not contain or transmit design information and is not meant to be used for performing engineering assessments, evaluations, or examinations.

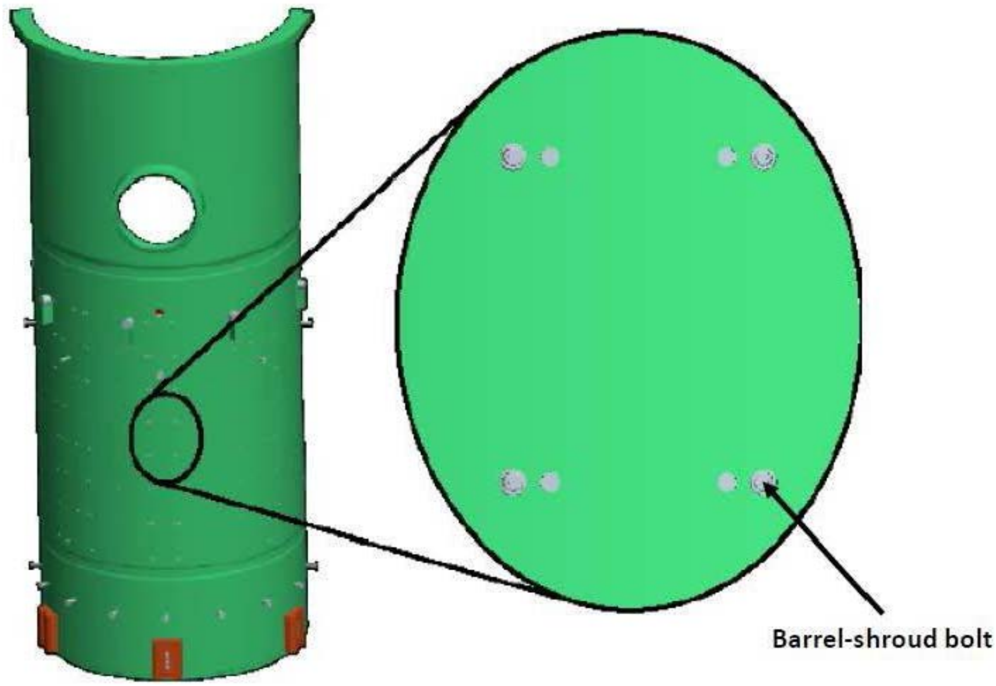

Figure 4-34
Typical CE-Design Core Shroud Assembly (Bolted) Barrel Shroud Bolts

Note: Figure intended to represent the component features and relative position to other portions of the structure and system. Figure does not contain or transmit design information and is not meant to be used for performing engineering assessments, evaluations, or examinations.

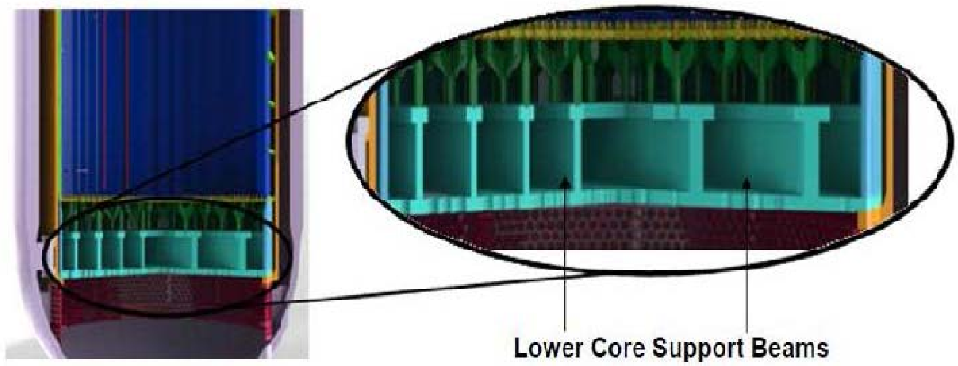

*Figure 4-35**
Typical CE-Design Lower Support Structure Lower Core Support Beams

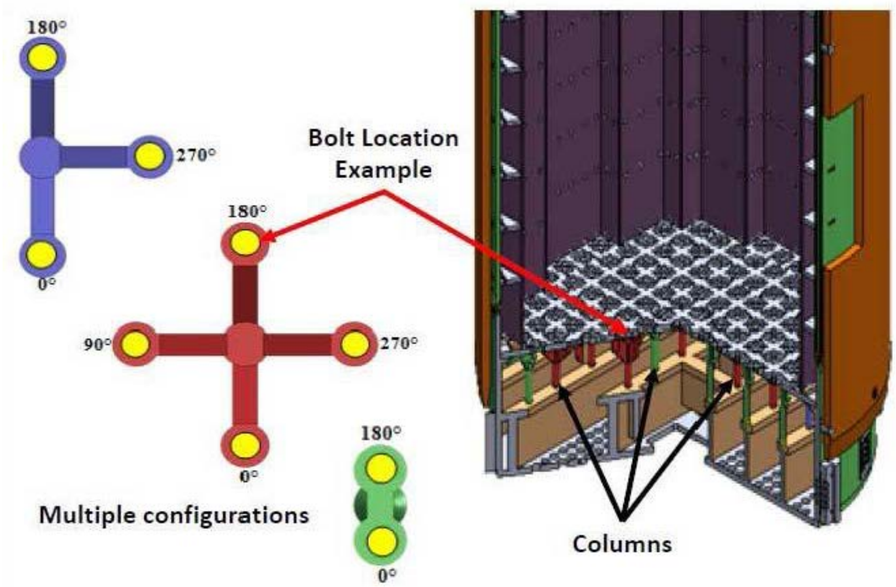

*Figure 4-36**
Typical CE-Design Core Shroud Assembly (Bolted) Core Support Column Bolts

* Note: Figure intended to represent the component features and relative position to other portions of the structure and system. Figure does not contain or transmit design information and is not meant to be used for performing engineering assessments, evaluations, or examinations.

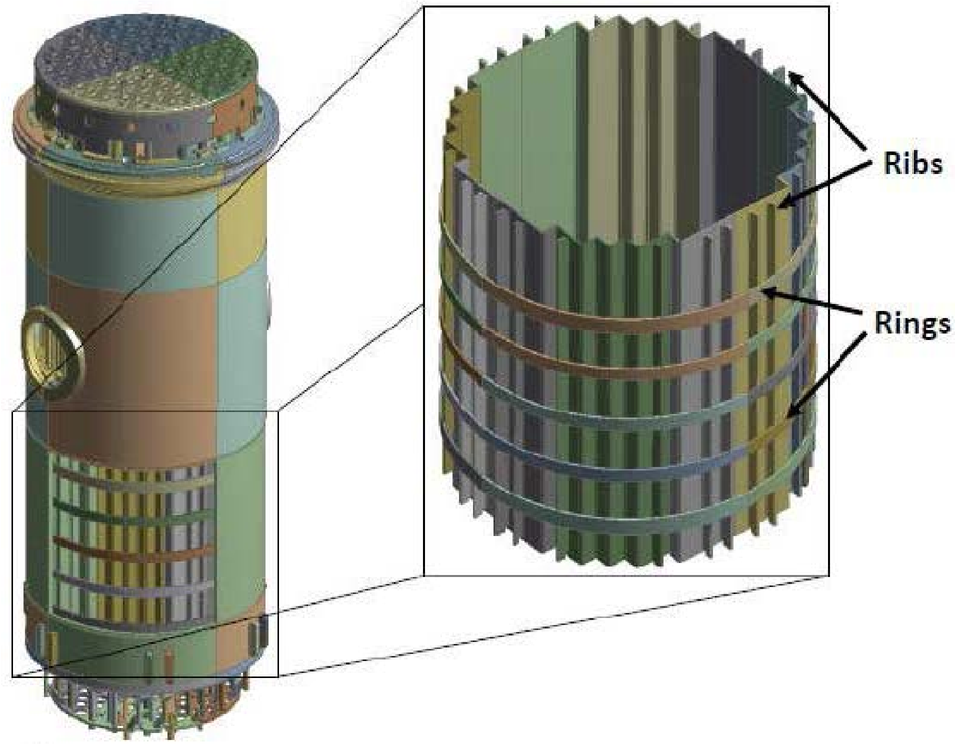

Figure 4-37
Typical CE-Design Core Shroud Assembly (Welded Full-Height) Core Shroud Assembly

Note: Figure intended to represent the component features and relative position to other portions of the structure and system. Figure does not contain or transmit design information and is not meant to be used for performing engineering assessments, evaluations, or examinations.

Section 5: Examination Acceptance Criteria and Expansion Criteria

The purpose of this section is to provide both examination acceptance criteria for conditions detected as a result of the examination requirements in Section 4, Tables 4-1 through 4-6, as well as criteria for expanding examinations to the Expansion components when warranted by the level of degradation detected in the Primary components.

Examination acceptance criteria identify the visual examination relevant condition(s) or signal-based level or relevance of an indication that requires formal disposition for acceptability. Based on the identified condition, and supplemental examinations if required, the disposition process results in an evaluation and determination of whether to accept the condition until the next examination or repair or replace the item. An acceptable disposition process is described in Section 6 and in WCAP-17096-NP-A [26]. Section 5.1 provides a discussion of relevant conditions applicable to the visual examination methods and of relevant indications applicable to the volumetric examinations employed in the guidelines. Section 5.2 provides examination acceptance criteria for physical measurements. These criteria are contained in Tables 5-1, 5-2, and 5-3 for B&W, CE, and Westinghouse plants, respectively.

Additionally, Tables 5-1, 5-2, and 5-3 contain expansion criteria for B&W, CE, and Westinghouse plants, respectively. Expansion criteria are intended to form the basis for decisions about expanding the set of components selected for examination or other aging management activity, to determine whether the level of degradation represented by the detected conditions has extended to other components judged to be less affected by the degradation.

The component population for expansion criteria is a population which has received adequate service time to experience the relevant aging degradation mechanism. For example, if a plant has replaced a partial number of relevant components, and the replacement components have not received adequate operating time by calendar year or EFPY to experience degradation as specified in Table 4-1, 4-2, or 4-3, then only the original population of components would be considered toward expansion criteria.

Table 5-1
B&W Plants Examination Acceptance and Expansion Criteria

Primary Item	Applicability	Primary Item Examination Acceptance Criteria (Notes 1, 2)	Expansion Link(s)	Expansion Criteria (Note 2)	Expansion Item Examination Acceptance Criteria
B1.Plenum Cover Assembly & Core Support Shield Assembly Plenum cover weldment rib pads Plenum cover support flange Plenum cover support ring CSS top flange	All plants	One-time physical measurement. In addition, a subsequent visual (VT-3) examination is conducted for these items. The measured differential height from the top of the plenum rib pads and plenum cover support ring to the vessel seating surface shall average less than 0.004 inches compared to the as-built condition. The specific relevant condition for the subsequent VT-3 of these items is a) evidence of a general polished area over the plenum cover support ring and plenum cover weldment rib pad region and a smeared image of the RV closure head contact region, or b) observance of an interrupted ring in the circumferential direction on the topside of the CSS top flange contact region.	None	N/A	N/A
B2.Control Rod Guide Tube Assembly CRGT spacer castings	All plants	Visual (VT-3) examination. The specific relevant condition is evidence of fractured spacers, missing screws, or any other anomalies near the threaded region.	B2.1.Vent valve bodies	Confirmed evidence of relevant conditions in two or more CRGT spacer castings shall require that the VT-3 examination be expanded to the vent valve bodies by the completion of the next refueling outage.	The specific relevant condition for the VT-3 of the expansion vent valve bodies is evidence of surface irregularities (including damaged, grossly cracked, or missing portions) to the vent valve bodies or visible damage to the jackscrews or locking devices.

Table 5-1 (continued)
B&W Plants Examination Acceptance and Expansion Criteria

Primary Item	Applicability	Primary Item Examination Acceptance Criteria (Notes 1, 2)	Expansion Link(s)	Expansion Criteria	Expansion Item Examination Acceptance Criteria
B3.Vent Valve Assembly Vent valve top retaining ring Vent valve bottom retaining ring	All plants	Visual (VT-3) examination. The specific relevant condition is evidence of damaged or fractured retaining ring material in the flanged region of the retaining ring, evidence of missing material, or evidence that the threaded connection has failed.	None	N/A	N/A
B4.Vent Valve Assembly Original locking devices (pressure plate, spring retainer, spring, U-cover)	Note 3	Visual (VT-3) examination. The specific relevant condition is evidence of damage or wear to the U-cover; misalignment of the pressure plate with the jackscrew, U-cover, or spring retainer; damage to the pressure plate or spring retainer; or the jackscrew out of the design configuration. Other specific relevant conditions are evidence that the valve is not symmetrical in the mounting ring or the jackscrew thread extensions from the lower retaining ring threaded flange are unequal.	None	N/A	N/A
B5.Vent Valve Assembly Original locking devices (key ring, pin)	Note 3	Visual (VT-3) examination. The key ring and pin are inaccessible items; therefore, the specific relevant condition is evidence that the valve is not symmetrical in the mounting ring or the jackscrew thread extensions from the lower retaining ring threaded flange are unequal.	None	N/A	N/A

Table 5-1 (continued)
B&W Plants Examination Acceptance and Expansion Criteria

Primary Item	Applicability	Primary Item Examination Acceptance Criteria (Notes 1, 2)	Expansion Link(s)	Expansion Criteria	Expansion Item Examination Acceptance Criteria
B6.Vent Valve Assembly Modified locking devices (bolt locking cup, jackscrew locking cup, bolted block)	Note 3	Visual (VT-3) examination. The specific relevant condition is evidence of fractured or missing locking cups and welds; a torn crimp or visual evidence that the crimp and sleeve slot are not aligned or engaged; or a fractured or missing bolted block. Other specific relevant conditions are missing cap screws associated with the crimped locking cups or the jackscrew thread extensions from the lower retaining ring threaded flange are unequal.	None	N/A	N/A

Table 5-1 (continued)
B&W Plants Examination Acceptance and Expansion Criteria

Primary Item	Applicability	Primary Item Examination Acceptance Criteria (Notes 1, 2)	Expansion Link(s)	Expansion Criteria	Expansion Item Examination Acceptance Criteria
B7.Core Support Shield Assembly Upper core barrel (UCB) bolts and their locking devices	All plants	1) Volumetric (UT) examination of the UCB bolts. The examination acceptance criteria shall be established as part of the examination technical justification. 2) Visual (VT-3) examination of the UCB bolt locking devices. The specific relevant condition is loss of material, damaged or distorted or missing bolt locking devices or welds.	B7.1.UTS bolts and B8.1.LTS bolts or studs/nuts and their locking devices B7.2.SSHT bolts and their locking devices (DB only)	1) Confirmed unacceptable indications exceeding 10% of the UCB bolts (including previously failed/removed bolts) shall require that the UT examination be expanded by the completion of the next refueling outage to include: *For all plants* 100% of the accessible UTS bolts and 100% of the accessible LTS bolts or studs/nuts, *Additionally for DB* 100% of the accessible SSHT bolts. 2) Confirmed evidence of relevant conditions exceeding 10% of the UCB bolt locking devices (including those previously failed/removed) shall require that the VT-3 examination be expanded by the completion of the next refueling outage to include: *For all plants* 100% of the accessible UTS bolt and 100% of the accessible LTS bolt or stud/nut locking devices. *Additionally for DB* 100% of the accessible SSHT bolt locking devices.	1) The examination acceptance criteria for the UT of the expansion bolting or studs/nuts shall be established as part of the examination technical justification. 2) The specific relevant condition for the VT-3 of the expansion locking devices is evidence of broken, separated, or missing bolt or stud/nut locking devices or welds.

Table 5-1 (continued)
B&W Plants Examination Acceptance and Expansion Criteria

Primary Item	Applicability	Primary Item Examination Acceptance Criteria (Notes 1, 2)	Expansion Link(s)	Expansion Criteria	Expansion Item Examination Acceptance Criteria
B8.Core Barrel Assembly Lower core barrel (LCB) bolts and their locking devices	All plants	1) Volumetric (UT) examination of the LCB bolts. The examination acceptance criteria shall be established as part of the examination technical justification. 2) Visual (VT-3) examination of the LCB bolt locking devices. The specific relevant condition is loss of material, damaged or distorted or missing bolt locking devices or welds.	B7.1.UTS bolts and B8.1.LTS bolts or studs/nuts and their locking devices B7.2.SSHT bolts and their locking devices (DB only)	1) Confirmed unacceptable indications exceeding 10% of the LCB bolts (including previously failed/removed bolts) shall require that the UT examination be expanded by the completion of the next refueling outage to include: *For all plants* 100% of the accessible UTS bolts and 100% of the accessible LTS bolts or studs/nuts, *Additionally for DB* 100% of the accessible SSHT bolts. 2) Confirmed evidence of relevant conditions exceeding 10% of the LCB bolt locking devices (including those previously failed/removed) shall require that the VT-3 examination be expanded by the completion of the next refueling outage to include: *For all plants* 100% of the accessible UTS bolts and 100% of the accessible LTS bolt or stud/nut locking devices. *Additionally for DB* 100% of the accessible SSHT bolt locking devices.	1) The examination acceptance criteria for the UT of the expansion bolting or studs/nuts shall be established as part of the examination technical justification. 2) The specific relevant condition for the VT-3 of the expansion locking devices is evidence of broken, separated, or missing bolt or stud/nut locking devices or welds.

Table 5-1 (continued)
B&W Plants Examination Acceptance and Expansion Criteria

Primary Item	Applicability	Primary Item Examination Acceptance Criteria (Notes 1, 2)	Expansion Link(s)	Expansion Criteria	Expansion Item Examination Acceptance Criteria
B9.Core Barrel Assembly Baffle-to-former bolts	All plants	Volumetric (UT) examination. The examination acceptance criteria shall be established as part of the examination technical justification.	B9.1.Baffle-to-baffle bolts B9.2.Core barrel-to-former bolts	Confirmed unacceptable indications in greater than or equal to 5% of the baffle-to-former bolts (including previously failed/removed bolts) shall require an evaluation of the baffle-to-baffle bolts and core barrel-to-former bolts by the completion of the next refueling outage. The evaluation shall also assess functionality of the core barrel assembly with aging degradation of the baffle-to-baffle bolts and core barrel-to-former bolts for the purpose of determining continued operation or replacement.	N/A
B10.Core Barrel Assembly Baffle plates	All plants	Visual (VT-3) examination. The specific relevant condition is readily detectable cracking in the baffle plates.	B10.2.Former plates B10.1.Core barrel cylinder (including vertical and circumferential seam welds) B10.3.Lower grid rib section	Gross cracking (if confirmed) within one inch of a bolt or flow hole location in the baffle plates shall require: a) An evaluation of the former plates and the core barrel cylinder for the purpose of determining continued operation or repair/replacement by the completion of the next refueling outage. Alternatively, repair/replacement activities may be initiated based on results of a best effort former plate and core barrel cylinder examination. b) That the VT-3 examination be expanded by the completion of the next refueling outage to include 100% of the accessible portions of the lower grid rib section heat-affected zones adjacent to the IMI guide tube spider-to-lower grid rib section welds.	N/A

Table 5-1 (continued)
B&W Plants Examination Acceptance and Expansion Criteria

Primary Item	Applicability	Primary Item Examination Acceptance Criteria (Notes 1, 2)	Expansion Link(s)	Expansion Criteria	Expansion Item Examination Acceptance Criteria
B11.Core Barrel Assembly Locking devices, including locking welds, of baffle-to-former bolts and internal baffle-to-baffle bolts	All plants	Visual (VT-3) examination. The specific relevant condition is evidence of broken, separated, or missing bolt locking devices or welds.	B11.1.Locking devices, including locking welds, for the external baffle-to-baffle bolts and core barrel-to-former bolts	Confirmed relevant aging degradation conditions in greater than or equal to 1% of the baffle-to-former and internal baffle-to-baffle bolt locking devices and locking welds (including those previously failed/removed) shall require an evaluation of the external baffle-to-baffle and core barrel-to-former bolt locking devices and welds for the purpose of determining continued operation or replacement by the completion of the next refueling outage.	N/A
B14.Lower Grid Assembly Shock pad bolts and their locking devices	TMI-1	1) Volumetric (UT) examination of the shock pad bolts. The examination acceptance criteria shall be established as part of the examination technical justification. 2) Visual (VT-3) examination of the shock pad bolt locking devices. The specific relevant condition is loss of material, damaged or distorted or missing bolt locking devices or welds.	B7.1.UTS bolts and B8.1.LTS bolts and their locking devices	1) Confirmed unacceptable indications exceeding 10% of the shock pad bolts (including previously failed/removed bolts) shall require that the UT examination be expanded by the completion of the next refueling outage to include 100% of the accessible UTS bolts and 100% of the accessible LTS bolts. 2) Confirmed evidence of relevant conditions exceeding 10% of the shock pad bolt locking devices (including those previously failed/removed) shall require that the VT-3 examination be expanded by the completion of the next refueling outage to include 100% of the accessible UTS bolts and 100% of the accessible LTS bolt locking devices.	1) The examination acceptance criteria for the UT of the expansion bolting shall be established as part of the examination technical justification. 2) The specific relevant condition for the VT-3 of the expansion locking devices is evidence of broken, separated, or missing bolt locking devices or welds.

Table 5-1 (continued)
B&W Plants Examination Acceptance and Expansion Criteria

Primary Item	Applicability	Primary Item Examination Acceptance Criteria (Notes 1, 2)	Expansion Link(s)	Expansion Criteria	Expansion Item Examination Acceptance Criteria
B13.Lower Grid Assembly Alloy X-750 dowel-to-guide block welds	All plants (except DB) (Note 4)	Visual (VT-3) examination. The specific relevant condition is separated or missing locking welds, or missing dowels.	Alloy X-750 dowel locking welds to the B13.1.upper and B13.2.lower grid fuel assembly support pads	Confirmed evidence of relevant conditions at two or more locations shall require that the VT-3 examination be expanded to include 100% of the Alloy X-750 dowel locking welds to the upper and lower grid fuel assembly support pads by the completion of the next refueling outage.	The specific relevant condition for the VT-3 of the expansion dowel locking weld is separated or missing locking welds, or missing dowels.
B12.Flow Distributor Assembly Flow distributor (FD) bolts and their locking devices	All plants	1) Volumetric (UT) examination of the FD bolts. The examination acceptance criteria shall be established as part of the examination technical justification. 2) Visual (VT-3) examination of the FD bolt locking devices. The specific relevant condition is loss of material, damaged or distorted or missing bolt locking devices or welds.	B7.1.UTS bolts and B8.1.LTS bolts or studs/nuts and their locking devices B7.2.SSHT bolts and their locking devices (DB only)	1) Confirmed unacceptable indications exceeding 10% of the FD bolts (including previously failed/removed bolts) shall require that the UT examination be expanded by the completion of the next refueling outage to include: _For all plants_ 100% of the accessible UTS bolts and 100% of the accessible LTS bolts or studs/nuts, _Additionally for DB_ 100% of the accessible SSHT bolts. 2) Confirmed evidence of relevant conditions exceeding 10% of the FD bolt locking devices (including those previously failed/removed) shall require that the VT-3 examination be expanded by the completion of the next refueling outage to include: _For all plants_ 100% of the accessible UTS bolts and 100% of the accessible LTS bolt or stud/nut locking devices. _Additionally for DB,_ 100% of the accessible SSHT bolt locking devices.	1) The examination acceptance criteria for the UT of the expansion bolting or studs/nuts shall be established as part of the examination technical justification. 2) The specific relevant condition for the VT-3 of the expansion locking devices is evidence of broken, separated, or missing bolt or stud/nut locking devices or welds.

Table 5-1 (continued)
B&W Plants Examination Acceptance and Expansion Criteria

Primary Item	Applicability	Primary Item Examination Acceptance Criteria (Notes 1, 2)	Expansion Link(s)	Expansion Criteria	Expansion Item Examination Acceptance Criteria
B15.Incore Monitoring Instrumentation (IMI) Guide Tube Assembly IMI guide tube spiders and the IMI guide tube spider-to-lower grid rib section welds	All plants	1) Visual (VT-3) examination. The specific relevant conditions for the IMI guide tube spiders are fractured or missing spider arms or a spider arm that does not align with the lower grid fuel assembly support pad screw (as viewed from the top). 2) Visual (VT-3) examination. The specific relevant conditions for the IMI spider-to-lower grid rib section welds are separated or missing welds.	B15.1.Lower grid fuel assembly support pad component items: pad, pad-to-rib section welds, Alloy X-750 dowel, cap screw, and their locking welds	Confirmed evidence of relevant conditions at two or more IMI guide tube spider locations or IMI guide tube spider-to-lower grid rib section welds shall require that the VT-3 examination be expanded to include accessible surfaces of the pads, dowels, cap screws, and associated welds for 100% of the lower fuel assembly support pads by the completion of the next refueling outage.	The specific relevant conditions for the VT-3 of the lower grid fuel assembly support pad component items (pads, pad-to-rib section welds, Alloy X-750 dowels, cap screws, and their locking welds) are separated or missing welds, missing support pads, dowels, cap screws and locking welds, or misalignment of the support pads.

Notes to Table 5-1:

1. The examination acceptance criterion for visual examination is the absence of the specified relevant condition(s).
2. Refer to MRP-231 Section 3.3 for additional details on examination acceptance and expansion criteria.
3. As of May 2014, TMI-1 and DB have been verified not to have modified vent valve assembly locking devices, while ONS-1, ONS-2, ONS-3, and ANO-1 have both original and modified vent valve assembly locking devices installed.
4. For the alternate configuration at DB, the difference in configuration shall be addressed on a unit-specific basis.

Table 5-2
CE Plants Examination Acceptance and Expansion Criteria

Primary Item	Applicability	Examination Acceptance Criteria (Note 1)	Expansion Link(s)	Expansion Criteria	Expansion Item Examination Acceptance Criteria
C1.Core Shroud Assembly (Bolted) Core shroud bolts	Bolted plant designs	Volumetric (UT) examination. Detection of a flaw, as characterized by the examination technical justification, shall be cause for rejection of the bolt. The examination acceptance criteria for the UT of the core shroud bolts shall be established as part of the examination technical justification.	a. C1.1.Core support column bolts b. C1.2.Barrel-shroud bolts	a. Confirmation that >5% of the core shroud bolts in the four plates at the largest distance from the core contain unacceptable indications shall require inspection of the lower support column bolts within the next 3 refueling cycles. b. Confirmation that >5% of the core support column bolts contain unacceptable indications shall require inspection of the barrel-shroud bolts within the next 3 refueling cycles.	Detection of a flaw, as characterized by the examination technical justification, shall be cause for rejection of the bolt. The examination acceptance criteria for the UT of the core support column bolts and barrel-shroud bolts shall be established as part of the examination technical justification.
C2.Core Shroud Assembly (Welded) Core shroud plate-former plate weld	Plant designs with core shrouds assembled in two vertical sections	Visual (EVT-1) examination. The specific relevant condition is a detectable crack-like surface indication.	C2.1.Remaining axial welds	Confirmation that a surface-breaking indication > 2 inches in length has been detected and sized in the core shroud plate-former plate weld at the core shroud re-entrant corners (as visible from the core side of the shroud), within 6 inches of the central flange and horizontal stiffeners, shall require inspection of all remaining axial welds by the completion of the next refueling outage.	The specific relevant condition is a detectable crack-like surface indication.
C3.Core Shroud Assembly (Welded) Shroud plates	Only plants with welded core shrouds assembled with full-height shroud plates	Visual (EVT-1) examination. The specific relevant condition is a detectable crack-like surface indication.	a. C3.1.Remaining axial welds b. C3.2.Ribs and rings	a. Confirmation that a surface-breaking indication > 2 inches in length has been detected and sized in the axial weld seams at the core shroud re-entrant corners at the core mid-plane shall require inspection of all remaining axial welds by the completion of the next refueling outage. b. If extensive cracking is detected in the remaining axial welds, an evaluation must be completed that justifies that the aging effects do not adversely affect the ability of the ribs and rings to perform their function. This evaluation must be finished by the completion of the next refueling outage. Alternatively, the component may be replaced.	The specific relevant condition is a detectable crack-like surface indication.

Table 5-2 (continued)
CE Plants Examination Acceptance and Expansion Criteria

Primary Item	Applicability	Examination Acceptance Criteria(Note 1)	Expansion Link(s)	Expansion Criteria	Expansion Item Examination Acceptance Criteria
C4.Core Shroud Assembly (Bolted) Assembly	Bolted plant designs	Visual (VT-3) examination. The specific relevant conditions are evidence of abnormal interaction with fuel assemblies, gaps along high fluence shroud plate joints, and vertical displacement of shroud plates near high fluence joints.	None	N/A	N/A
C4a.Core Shroud Assembly (Welded) Assembly	Plant designs with core shrouds assembled in two vertical sections	Visual (VT-1) examination. The specific relevant condition is evidence of physical separation between the upper and lower core shroud sections.	None	N/A	N/A
C5.Core Support Barrel Assembly Upper Flange Weld (UFW)	All plants	Visual (EVT-1) examination. The specific relevant condition is a detectable crack-like surface indication.	C5.2.Upper Girth Weld (UGW) C5.1.Lower Girth Weld (LGW) C5.3.Upper Axial Weld (UAW) C5.4.Lower core support beams	a. The confirmed detection and sizing of a surface-breaking indication with a length greater than two inches in the UFW shall require that the inspection be expanded to include the UGW, LGW, and UAW by the completion of the next refueling outage. b. The confirmed detection and sizing of a surface-breaking indication with a length greater than two inches in the UFW shall require inspection of the lower core support beams within the next three refueling outages.	The specific relevant condition is a detectable crack-like surface indication.

Table 5-2 (continued)
CE Plants Examination Acceptance and Expansion Criteria

Primary Item	Applicability	Examination Acceptance Criteria(Note 1)	Expansion Link(s)	Expansion Criteria	Expansion Item Examination Acceptance Criteria
C6.Core Support Barrel Assembly Middle Girth Weld (MGW)	All plants	Visual (EVT-1) examination. The specific relevant condition is a detectable crack-like surface indication.	C6.1.Middle Axial Weld (MAW) C6.2.Lower Axial Weld (LAW) C6.3.Core Support Columns	The confirmed detection and sizing of a surface-breaking indication with a length greater than two inches in the MGW shall require that the inspection be expanded to include the MAW and LAW by the completion of the next refueling outage. The confirmed detection of a surface-breaking linear indication in the MGW shall require examination of 25% (of the total of both visible and non-visible as seen from above the core support plate) of the core support column assemblies by the completion of the next refueling outage. Plants with full-height bolted core shroud plates: The confirmed detection of missing or separated welds in a core support column or fractured, misaligned, or missing core support columns shall require examination of 100% of the accessible uninspected core support column assemblies using a VT-3 examination from above the core support plate (minimum of 75% of the total population of core support column assemblies) during the same outage. Plants with core shrouds assembled in two vertical sections: The confirmed detection of a relevant disruption of discontinuity in the surface of a core support column weld shall require examination of 100% of the accessible uninspected core support column welds from the top side of the core support plate (minimum of 75% of the total population of core support column welds) during the same outage.	The specific relevant condition for the expansion lower cylinder axial welds is a detectable crack-like surface indication. The specific relevant condition for the core support column welds is a disruption or discontinuity in the surface of the weld. The specific relevant condition for the core support columns viewed from above the core support plate is missing or separated welds, or fractured, misaligned, or missing columns.

Table 5-2 (continued)
CE Plants Examination Acceptance and Expansion Criteria

Primary Item	Applicability	Examination Acceptance Criteria(Note 1)	Expansion Link(s)	Expansion Criteria	Expansion Item Examination Acceptance Criteria
C7.Core Support Barrel Assembly CSB Flexure weld (CSBFW)	All plants with welded core shrouds	Visual (EVT-1) examination. The specific relevant condition is a detectable crack-like indication.	None	N/A	N/A
C9.Lower Support Structure Core support plate	All plants with a core support plate	Visual (EVT-1) examination. The specific relevant condition is a detectable crack-like surface indication.	None	N/A	N/A
C10.Upper Internals Assembly Fuel alignment plate	Only plants with welded core shrouds assembled with full-height shroud plates	Visual (EVT-1) examination. The specific relevant condition is a detectable crack-like surface indication.	None	N/A	N/A
C11.Control Element Assembly Instrument guide tubes	All plants with instrument guide tubes in the CEA shroud assembly	Visual (VT-3) examination. The specific relevant conditions are missing supports or separation at the welded joint between the tubes and the supports.	C11.1.Remaining instrument guide tubes within the CEA shroud assemblies	Confirmed evidence of missing supports or separation at the welded joint between the tubes and supports shall require the inspection coverage to be expanded to the remaining instrument tubes within the CEA shroud assemblies by completion of the next refueling outage.	The specific relevant conditions are missing supports or separation at the welded joint between the tubes and the supports.
C12.Lower Support Structure Deep beams	Only plants with welded core shrouds assembled with full-height shroud plates	Visual (EVT-1) examination. The specific relevant condition is a detectable crack-like indication.	Remaining deep beams	Confirmed evidence of a detectable crack-like indication in one or more of the deep beams examined in the 25% population selected for the Primary component inspection shall require the inspection coverage to be expanded to the remaining deep beams by the completion of the next refueling outage.	The specific relevant condition is a detectable crack-like indication.

Note to Table 5-2:

1. The examination acceptance criterion for visual examination is the absence of the specified relevant condition(s).

Table 5-3
Westinghouse Plants Examination Acceptance and Expansion Criteria

Primary Item	Applicability	Examination Acceptance Criteria (Note 1)	Expansion Link(s)	Expansion Criteria	Expansion Item Examination Acceptance Criteria
W1.Control Rod Guide Tube Assembly Guide plates (cards)	All plants	Per the requirements of WCAP-17451-P The specific relevant condition is wear that could lead to loss of control rod alignment and impede control assembly insertion.	None	N/A	Per WCAP-17451-P [37].
W2.Control Rod Guide Tube Assembly Lower flange welds	All plants	Enhanced visual (EVT-1) examination. The specific relevant condition is a detectable crack-like surface indication.	W2.1.Remaining accessible CRGT lower flange welds W2.2.Bottom-mounted instrumentation (BMI) column bodies	Confirmation of surface-breaking indications in two or more CRGT lower flange welds shall require visual (EVT-1) examination of the remaining accessible CRGT lower flange welds and visual (VT-3) examination of BMI column bodies by the completion of the next refueling outage.	For BMI column bodies, the specific relevant condition for the VT-3 examination is completely fractured column bodies.
W3.Core Barrel Assembly Upper flange weld (UFW)	All plants	Periodic enhanced visual (EVT-1) examination. The specific relevant condition is a detectable crack-like surface indication.	W3.1.Upper girth weld (UGW) W3.3.Lower flange weld (LFW) (Note 2) W3.2.Upper axial welds (UAW) W3.4.Lower support forging/casting	a. The confirmed detection and sizing of a surface-breaking indication with a length greater than two inches in the UFW shall require that the inspection be expanded to include the UGW and LFW by the completion of the next refueling outage. b. The confirmed detection and sizing of a surface breaking indication with a length greater than two inches in either the UGW or LFW shall require that the inspection be expanded to include the UAW by the completion of the next refueling outage. c. The confirmed detection of a surface-breaking indication with a length greater than two inches in the LFW shall require the inspection of the lower support forging or casting (25% of the non-core side surface) within the next three refueling outages. If an indication is found in this inspection of the lower support forging or casting, the examination coverage shall be expanded to 100% of the accessible surface of the non-core side surface of the lower support forging or casting during the same refueling outage.	The specific relevant condition for the expansion core barrel welds (UGW, LFW, UAW) and lower support forging or casting examinations is a detectable crack-like surface indication.

Table 5-3 (continued)
Westinghouse Plants Examination Acceptance and Expansion Criteria

Primary Item	Applicability	Examination Acceptance Criteria (Note 1)	Expansion Link(s)	Expansion Criteria	Expansion Item Examination Acceptance Criteria
W4.Core Barrel Assembly Lower girth weld (LGW)	All plants	Periodic enhanced visual (EVT-1) examination. The specific relevant condition is a detectable crack-like surface indication.	W4.1.Upper core plate W4.4.Lower support column bodies (cast and non-cast) W4.2.Middle axial welds (MAW) W4.3.Lower axial welds (LAW)	a. The confirmed detection and sizing of a surface-breaking indication with a length greater than two inches in the LGW shall require inspection of the upper core plate (25% of the core-side surface) within the next three refueling outages. If an indication is found in this inspection of the upper core plate, the examination coverage shall be expanded to 100% of the accessible surface of the core-side surface of the upper core plate during the same refueling outage. b. The confirmed detection and sizing of a surface-breaking indication with a length greater than two inches in the LGW shall require inspection of the lower support column bodies (cast and non-cast) within the next three refueling outages. The confirmed detection of fractured, misaligned, or missing lower support columns shall require examination of 100% of the accessible uninspected lower support column assemblies using a VT-3 examination from above the lower core plate (minimum of 75% of the total population of lower support column assemblies) during the same outage. c. The confirmed detection and sizing of a surface-breaking indication with a length greater than two inches in the LGW shall require that the inspections be expanded to include the lower core barrel cylinder axial welds by the completion of the next refueling outage.	a. The specific relevant conditions for the inspection of the upper core plate are broken or missing parts of the plate. b. The specific relevant conditions for the inspection of the lower support column bodies (cast and non-cast) are fractured, misaligned, or missing columns. c. The specific relevant condition for the expansion MAW and LAW inspections is a detectable crack-like surface indication.

Table 5-3 (continued)
Westinghouse Plants Examination Acceptance and Expansion Criteria

Primary Item	Applicability	Examination Acceptance Criteria (Note 1)	Expansion Link(s)	Expansion Criteria	Expansion Item Examination Acceptance Criteria
W5.Baffle-Former Assembly Baffle-edge bolts	All plants with baffle-edge bolts	Visual (VT-3) examination. The specific relevant conditions are missing or broken locking devices, cracked/failed or missing bolts, and protrusion of bolt heads.	None	N/A	N/A
W6.Baffle-Former Assembly Baffle-former bolts	All plants	Volumetric (UT) examination. The examination acceptance criteria for the UT of the baffle-former bolts shall be established as part of the examination technical justification.	W6.2.Lower support column bolts W6.1.Barrel-former bolts	a. Confirmation that more than 5% of the baffle-former bolts actually examined on the four baffle plates at the largest distance from the core (presumed to be the lowest dose locations) contain unacceptable indications shall require inspection of the lower support column bolts within the next three fuel cycles. b. Confirmation that more than 5% of the lower support column bolts actually examined contain unacceptable indications shall require inspection of the barrel-former bolts within three refueling cycles.	The examination acceptance criteria for the UT of the lower support column bolts and the barrel-former bolts shall be established as part of the examination technical justification.

Table 5-3 (continued)
Westinghouse Plants Examination Acceptance and Expansion Criteria

Primary Item	Applicability	Examination Acceptance Criteria (Note 1)	Expansion Link(s)	Expansion Criteria	Expansion Item Examination Acceptance Criteria
W7.Baffle-Former Assembly Assembly (Includes: Baffle plates, baffle edge bolts, corner bolts, and indirect effects of void swelling in former plates)	All plants	Visual (VT-3) examination. The specific relevant conditions are evidence of abnormal interaction with fuel assemblies, gaps along high fluence baffle plate joints, vertical displacement of baffle plates near high fluence joints, or more than 2 broken or damaged edge bolt locking systems along high fluence baffle plate joints.	None	N/A	N/A
W8.Alignment and Interfacing Components Internals hold down spring	All plants with 304 stainless steel hold down springs	Direct physical measurement of spring height. The examination acceptance criterion for this measurement is that the remaining compressible height of the spring shall provide hold-down forces within the plant-specific design tolerance.	None	N/A	N/A
W9.Thermal Shield Assembly Thermal shield flexures	All plants with thermal shields	Visual (VT-3) examination. The specific relevant conditions for thermal shield flexures are excessive wear, fracture, or complete separation.	None	N/A	N/A

Notes to Table 5-3:

1. The examination acceptance criterion for visual examination is the absence of the specified relevant condition(s).
2. The lower core barrel flange weld may alternatively be designated as the core barrel-to-support plate weld in some Westinghouse plant designs.

5.1 Examination Acceptance Criteria

5.1.1 Visual (VT-3) Examination

Visual (VT-3) examination has been determined to be an appropriate NDE method for the detection of general degradation conditions in many of the susceptible components. The ASME Code Section XI, Examination Category B-N-3 [2], provides a set of relevant conditions for the visual (VT-3) examination of removable core support structures in IWB-3520.2. These are:

1. Structural distortion or displacement of parts to the extent that component function may be impaired;
2. Loose, missing, cracked, or fractured parts, bolting, or fasteners;
3. Corrosion or erosion that reduces the nominal section thickness by more than 5%;
4. Wear of mating surfaces that may lead to loss of function; and
5. Structural degradation of interior attachments such that the original cross-sectional area is reduced more than 5%.

For components in the Existing Programs group, these general relevant conditions are sufficient. However, for components where visual (VT-3) is specified in the Primary or the Expansion group, more specific descriptions of the relevant conditions are provided in Tables 5-1, 5-2, and 5-3 for the benefit of the examiners. Typical examples are "fractured material" and "separated material." One or more of these specific relevant condition descriptions may be applicable to the Primary and Expansion components listed in Tables 5-1, 5-2, and 5-3.

The examination acceptance criteria for components requiring visual (VT-3) examination is thus the absence of the relevant condition(s) specified in Tables 5-1, 5-2, and 5-3.

The disposition can include a supplementary examination to further characterize the relevant condition, an engineering evaluation to show that the component is capable of continued operation with a known relevant condition, or repair/replacement to remediate the relevant condition.

5.1.2 Visual (VT-1) Examination

Visual (VT-1) examination is defined in the ASME Code Section XI [2] as an examination "conducted to detect discontinuities and imperfections on the surface of components, including such conditions as cracks, wear, corrosion, or erosion." For these guidelines VT-1 has only been selected to detect distortion as evidenced by small gaps between the upper-to-lower mating surfaces of CE welded core shrouds assembled in two vertical sections.

The examination acceptance criterion is thus the absence of the relevant condition of gaps that would be indicative of distortion from void swelling.

5.1.3 Enhanced Visual (EVT-1) Examination

Enhanced visual (EVT-1) examination has the same requirements as the ASME Code Section XI [2] visual (VT-1) examination, with additional requirements given in the Inspection Standard [3]. These enhancements are intended to improve the detection and characterization of discontinuities taking into account the remote visual aspect of PWR internals examinations. As a result, EVT-1 examinations are capable of detecting small surface breaking cracks and surface crack length sizing when used in conjunction with sizing aids (e.g., landmarks, ruler, and tape measure). EVT-1 examination has been selected to be the appropriate NDE method for detection of cracking in plates or their welded joints. Thus the relevant condition applied for EVT-1 examination is the same as found for cracking in Reference 2 which is crack-like surface breaking indications.

Therefore, until such time as generic engineering studies develop the basis by which a quantitative amount of degradation can be shown to be tolerable for the specific component, any relevant condition is to be dispositioned. In the interim, the examination acceptance criterion is thus the absence of any detectable surface breaking indication.

5.1.4 Surface Examination

Surface ET (eddy current) examination is specified as an alternative or as a supplement to visual examinations. No specific acceptance criteria for surface (ET) examination of PWR internals locations are provided in the ASME Code Section XI [2]. Since surface ET is employed as a signal-based examination, a technical justification per the Inspection Standard [3] provides the basis for detection and length sizing of surface-breaking or near-surface cracks. The signal-based relevant indication for surface (ET) examination is thus the same as the relevant condition for enhanced visual (EVT-1) examination. The acceptance criteria for enhanced visual (EVT-1) examinations in 5.1.3 (and accompanying entries in Tables 5-1, 5-2, and 5-3) are therefore applied when this method is used as an alternative or supplement to visual examination.

5.1.5 Volumetric Examination

The intent of volumetric examinations specified for bolts or pins in Section 4.3 of these I&E guidelines is to detect planar defects. No flaw sizing measurements are recorded or assumed in the acceptance or rejection of individual bolts or pins. Individual bolts or pins are accepted based on the detection of relevant indications established as part of the examination technical justification. When a relevant indication is detected in the cross-sectional area of the bolt or pin, it is assumed to be non-functional and the indication is recorded. A bolt or pin that passes the criterion of the examination is assumed to be functional.

Because of this pass/fail acceptance of individual bolts or pins, the examination acceptance criterion for volumetric (UT) examination of bolts and pins is based on a reliable detection of indications as established by the individual technical justification for the proposed examination. This is in keeping with current industry practice. For example, planar flaws on the order of 30% of the cross-sectional area have been demonstrated to be reliably detectable in previous bolt NDE technical justifications for baffle-former bolting.

Bolted and pinned assemblies are evaluated for acceptance based on meeting a specified number and distribution of functional bolts and pins. Evaluation options are discussed in Section 6.

5.2 Physical Measurements Examination Acceptance Criteria

Continued functionality can be confirmed by physical measurements where, for example, loss of material caused by wear, loss of pre-load of clamping force caused by various degradation mechanisms, or distortion/deflection caused by void swelling may occur. Where appropriate, these physical measurements are described in Section 4.2.5, with limits or tolerances applicable to the various designs. These measurements are to be compared to established limits or tolerances that are defined within a reasonable period of time prior to the measurements being taken. Satisfaction of limits on these physical measurements is intended to demonstrate that the affected components remain functional or they can continue in service for a determined period until the next set of physical measurements. If the limits are exceeded, it is recognized that the component(s) may still be acceptable as-is, but corrective action or evaluation for continued service is still required.

For B&W designs, the acceptable tolerance for the measured differential height from the top of the plenum rib pads to the vessel seating surface has been generically established and is provided in Table 5-1. For Westinghouse designs, tolerances are available on a design or plant-specific basis and thus are not provided generically in these guidelines. For CE designs, no physical measurements are specified.

5.3 Expansion Criteria

The criteria for expanding the scope of examination from the Primary components to their linked Expansion components are contained in Tables 5-1, 5-2, and 5-3 for B&W, CE, and Westinghouse plants, respectively.

Section 6: Evaluation Methodologies

There are various options available for the disposition of conditions detected during examinations (Section 4) that are unable to satisfy the examination acceptance criteria (Section 5). These options include, but are not limited to: (1) supplemental examinations, such as a surface examination, to supplement a visual (VT-1) or an enhanced visual (EVT-1) examination, to further characterize and potentially dispose of a detected condition; (2) engineering evaluation that demonstrates the acceptability of a detected condition; (3) repair to restore a component with a detected condition to acceptable status; or (4) replacement of a component with an unacceptable detected condition.

The first option involves the re-examination of a component with an unacceptable detected condition with an alternative examination method that has the potential capability to further define or confirm with greater precision the component physical condition. This additional characterization may enable the more precise character of that detected condition to be found acceptable for continued service. An example would be the volumetric (UT) examination to depth size a surface-breaking flaw detected by either visual (VT-1) or enhanced visual (EVT-1) examination.

The second option involves performing engineering evaluations. Acceptable methodologies range from the satisfaction of limit load requirements for the internals assembly or component cross section to the satisfaction of flaw stability requirements using either linear elastic fracture mechanics (LEFM) or elastic-plastic fracture mechanics (EPFM), depending upon applicability. The evaluation process depends upon the loading applied to the component, assembly, or system. In addition, flaw depth assumptions, in the absence of flaw depth sizing during examination, and flaw growth assumptions for subsequent operation until the next examination, must be considered. Justification for flaw evaluation fracture toughness limits must also be considered. This justification should include consideration of the potential impact of aging (e.g., thermal and irradiation embrittlement) on fracture toughness and flaw growth. In the report BWRVIP-100-A, EPRI developed an assessment of the fracture toughness of irradiated stainless steel for BWR core shrouds [19]. There is also some information related to fracture toughness data and models in various technical reports and industry literature [19-24], including the EPRI report MRP-211 [25]. EPRI also developed IASCC CGR models for both PWR and BWR internals [33] and these CGR models can be used in PWR internals evaluations. Design-specific or fleet-specific flaw handbooks may be used as an engineering evaluation tool. The NRC-approved report WCAP-17096-NP-A [26] provides

approved methods for evaluating examination results that do not meet the examination acceptance criteria in Section 5. The WCAP-17096-NP-A methods are to be followed in accordance with the Needed requirement of 7.5.

Other options include repair or replacement such as mechanical repairs or repairs using welding methods. It is recognized that implementation of these options are controlled by owner-specific and OEM-specific processes. As documented in MRP-379, EPRI developed a PWR irradiated materials welding guideline [32] that concluded that, in a typical PWR, a number of components can be repaired by welding with conventional techniques. Guidance presented in this MRP-379 report describes distances from landmark points in each reactor design where helium thresholds are anticipated to occur, and these distances can be used to evaluate weldability.

Section 7: Implementation Requirements

The purpose of this section is to summarize the implementation requirements of these guidelines. As stated previously, these guidelines do not reduce, alter, or otherwise affect current ASME B&PV Code Section XI or plant-specific licensing inservice inspection requirements. These guidelines do not apply to new plants beginning construction after calendar year 2007.

Utilities may elect to use this document to meet the requirement for an integrated plant assessment (IPA) under 10 CFR 54.21. If this approach is used, then the utility owners are to confirm that PWR internals components and materials within the scope of license renewal are consistent with MRP-189 and MRP-191. If components or materials are not consistent, then they must be dispositioned using the same process.

7.1 NEI 03-08 Implementation Protocol

These guidelines are a 'work product' of the EPRI MRP, an 'Issue Program (IP)' as defined in NEI 03-08 [1]. Appendix B to NEI 03-08, Implementation Protocol, defines the processes and expectations for implementing industry guidance issued under the Materials Initiative, and requires that IPs identify the specific implementation category for 'requirements' identified by guideline-type work products.

The three implementation categories described in NEI 03-08 are as follows:

- Mandatory – to be implemented at all plants where applicable;
- Needed – to be implemented wherever possible, but alternative approaches are acceptable; and
- Good Practice – implementation is expected to provide significant operational and reliability benefits, but the extent of use is at the discretion of the individual utility.

The following sections list or summarize the requirements contained in this document. A failure to meet a Needed or a Mandatory requirement is a deviation from the guidelines and a written justification for the deviation must be prepared and approved as described in Appendix B to NEI 03-08 [1]. A copy of the deviation is sent to the MRP so that improvements to the guidelines can be developed.

7.2 Aging Management Requirement

Mandatory: *Each commercial U.S. PWR unit shall develop, document and maintain an engineering program for management of aging of reactor internal components.*

Unless otherwise specified by the issuance letter for this guideline, updates of programs reflecting the requirements of this guideline may be in accordance with plant-specific utility procedures.

7.3 Reactor Internals Guidelines Implementation Requirement

Needed: *Each commercial U.S. PWR unit shall implement the requirements of Tables 4-1 through 4-9 and Tables 5-1 through 5-3 for the applicable design.*

Implementation means performance of examinations of applicable components within the timeframe specified in the applicable tables.

For units that have submitted an AMP to the regulator under MRP-227-A, and their period of extended operation begins no later than January 1, 2022, that MRP-227-A based program may be implemented as the baseline inspection without deviation from this Needed requirement. Requirements contained within the interim guidance letters applicable to that unit must also be incorporated in the plant program (e.g., MRP-2014-006, MRP-2013-023, MRP 2016-021, and MRP 2017-009). However, subsequent implementation shall be in accordance with revision 1-A of this guideline.

Updates of engineering programs for this revision of the guideline for each reactor shall be implemented by January 1, 2022; however, these NRC-endorsed guidelines may be implemented immediately, once published.

7.4 Examination Procedures Requirement

Needed: *Examinations specified in these guidelines shall comply with the Needed requirements in the MRP-228 Inspection Standard [3].*

7.5 Examination Results Requirement

Needed: *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).*

7.6 Results Reporting Requirement

Needed: *Each commercial U.S. PWR unit shall provide a summary report of all inspections and monitoring (including coverage(s) achieved and inspection limitations), items requiring evaluation, and new repairs to the MRP Program Manager within six (6) months of the completion of an outage during which PWR internals within the scope of MRP-227 are examined.*

This summary of the results will be compiled into an overall industry report which will track industry progress, aid in evaluation of significant issues, identify fleet trends and provide a determination of any needed revisions to these guidelines. The industry report will be updated biennially for the benefit of the fleet, the regulator, the PWROG and other industry stakeholders. This biennial report will serve to assist in review of operating experience, and required monitoring and trending for aging management programs established by the industry.

Section 8: References

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Appendix A: Reactor Internals Operational Experience

Note that in Revision 0 to MRP-227, Appendix A provided guidance for development of an AMP for PWR internals components. This guidance has been deleted from MPR-227. Guidance for AMP preparation may be found in AMP XI.M16A of NUREG-1801, Revision 2 (or subsequent revisions).

The following compilation of operational experience neither replaces efforts by licensees to review and document their plant-specific operating experience that may impact plant programs, nor does it preclude licensee participation in industry initiatives that perform these functions.

Commercial PWR vessel internals in the U.S. have experienced safe, relatively trouble-free operation. There have been no instances to date in which a PWR in the U.S. has posed a threat to public safety as a result of PWR internals material aging degradation. While relatively few incidents of PWR vessel internals aging degradation have been reported in operating U.S. commercial PWR plants, a summary of the current operating experience including international experience is useful for licensees developing aging management programs. This summary is organized first by the aging effect and subsequently by the age-related degradation mechanism leading to that effect. The operating experience includes findings made prior to and after the implementation of MRP-227.

Cracking

IGSCC – Three B&W-design PWR experienced internals bolt failures of the lower thermal shield bolts discovered during the 1981 and 1982 inservice inspections. The thermal shield bolt locking clips at these three plants were visually observed to be missing or loose. Subsequent examinations during 1982, 1983, and 1984 revealed bolt failures at four additional units. These failures included upper core barrel, lower core barrel, upper thermal shield, and surveillance specimen holder tube bolts. All of the affected fasteners were fabricated from Alloy A-286 ASTM A 453, Grade 660, Class A[7] or B[8] material. The results of an extensive evaluation program revealed the failure mechanism was predominantly due to an environmentally-assisted IGSCC mechanism.

[7] Solution anneal at 1650°F for 1 hour; oil or water quench; age at 1325°F for 16 hours; air cool.

[8] Solution anneal at 1800°F for 1 hour; oil or water quench; age at 1325°F for 16 hours; air cool.

However, for some bolts, there was evidence that fatigue was also a contributor, likely in the form of corrosion fatigue. The most recent examinations, per the guidelines in MRP-227-A, were completed in the 2007-2014 timeframe. Examinations of the upper core barrel, lower core barrel, and flow distributor bolts identified a small number of additional bolts with crack-like indications.

In general, the primary mechanism causing cracking and failure of the Alloy A-286 PWR internals bolts was IGSCC. All the failures occurred in the bolt head-to-shank fillet. Information Notice (IN) 90-68 provides information about IGSCC cracking in Alloy A-286 bolts used to hold the turning vanes to reactor coolant pumps at a foreign plant. The IN 90-68 document includes a general discussion of the problems experienced with cracking of Alloy A-286 bolting materials, including the problems identified with respect to B&W PWR internals bolting.

In addition, in 2012 as a result of impact damage to a B&W unit vent valve, an Alloy A-286 jackscrew normally in compression was subjected to a bending stress and cracked due to IGSCC.

In 2005, cracking of replacement core barrel-to-former plate bolts fabricated from cold-worked (CW) Type 316Ti stainless steel was observed in a German PWR by visual inspection. These bolts had replaced the original Alloy X-750 core barrel bolts in the late 1980s, which had exhibited failure due to PWSCC (described below). Subsequent UT inspection and failure analysis confirmed that the cracking was confined to the bolt head initiating from the bolt fillet transition, but the bolt threads and shank were free from cracking. The failure mechanism of the CW Type 316Ti stainless steel replacement core barrel bolts has been identified as IGSCC. To date, all known IGSCC failures of core barrel bolts have been limited to the original Alloy X-750 and the replacement CW Type 316Ti stainless steel in German PWRs.

PWSCC – Alloy X-750 has experienced numerous worldwide failures in the Westinghouse-designed PWR internals involving the control rod guide tube support pins (a.k.a., split pins). As noted in IN 82-29, these failures first appeared in Japan in the late 1970s. Split pin failures prompted investigations and modifications to manufacturing practices. The original heat treatment condition AH[9] of the age-hardenable material has shown the most susceptibility to PWSCC cracking. By the early 1980s, nearly all of the original design split pins had been replaced with the improved HTH[10] heat treatment condition.

In 1987, failures of Alloy X-750 HTH Condition control rod guide tube support pins in French PWRs occurred at much shorter times and lower stresses than expected. Foucault, et al., showed that these early failures were due to the surface condition of the pins. Any heat treatment after machining degrades the performance of Alloy X-750. The greatest resistance to IGSCC was found when

[9] Hot finish at 1800°F; equalize at 1625°F for 24 hours; age at 1300°F) for 20 hours; air cool.

[10] Hot roll to 35% min. reduction; solution anneal at 2025°F for 1 hour; rapid cool; age at 1300°F for 20 hours; air cool.

machining or polishing was performed after heat treatment, which removes an oxide layer from the surface of the material. Additional refinements have since been made to the manufacturing practices used to produce a newer version of Alloy X-750 HTH split pins.

After an extensive worldwide industry program to develop a material heat treatment for Alloy X-750 that would have maximum resistance to SCC, Westinghouse and utility customers conducted a campaign during the 1980s to replace guide tube support pins. Ultimately, Westinghouse developed a CW Type 316 stainless steel support pin as a replacement and a number of utilities have performed replacements with this design. A few utilities have opted to perform ultrasonic inspections rather than initiate wholesale replacements, while still other utilities have preferred to take no action at this time.

Alloy X-750, in a condition similar to AH, was used for the baffle-to-former plate bolts in the German Biblis-type reactors. After about four years of service, several bolts were found either cracked or severed. The cracking occurred in the bolt head-to-shank fillet area and was attributed to IGSCC (a.k.a., PWSCC in nickel-base materials). The bolt stress levels were reportedly at the yield strength of the material.

Failures have been attributed to three factors:

1. Heat treatment condition
2. High peak stresses
3. Surface damage due to fabrication processes

Failures of the lower radial support system (LRSS) Alloy X-750 clevis insert bolts were reported by one Westinghouse-designed plant in 2010. The lower clevis structure interfaces with the radial keyways on the core barrel to provide rotational alignment and lateral support for the lower internals. The Alloy X-750 bolting was used to fasten the Alloy 600 clevis inserts to the RV lugs. The failed clevis insert bolts were removed for maintenance. A metallurgical examination was performed on the removed bolts which showed the cause of failure to be PWSCC (Ref. Westinghouse TB-14-5). The clevis insert bolting had been heat treated using a low temperature process that has proven to be susceptible to PWSCC in the guide tube support pins. The relatively long time to failure in the clevis insert bolting may be attributed to the lower service temperature.

During 2012-2014 inspections of three B&W units with modified vent valve locking devices, crimped Alloy 600 locking cups were inspected and crack-like indications associated with some of the locking cup crimps were identified at two of the units.

IASCC – A considerable amount of PWR internals IASCC has been observed in European PWRs since the 1980s, with emphasis on cracking of baffle-former bolting. Ultrasonic (UT) testing of baffle-former bolts in six French PWRs discovered failure rates ranging from 1.2% to 11% of the 960 total bolts. For this reason, the U.S. PWR owners and operators began a program to inspect the

baffle-former bolting to determine whether similar problems might be expected in U.S. plants. One benefit of this program was the experience gained with the UT examination techniques used in the inspections. In addition, the industry began laboratory testing projects to gather the materials data necessary to support future inspections and evaluations.

As part of the U. S baffle-former bolt program, UT inspections were performed at two units with CW Type 316 stainless steel bolting (1998/1999). In one unit, 1086 of 1088 bolts were inspected with no indications. Two bolts could not be inspected due to accessibility and were replaced. In the second unit, all 1088 bolts were inspected, again with no indications. A proactive minimum bolt pattern replacement was performed at these plants (276 bolts for Unit 1 and 203 bolts for Unit 2). Bolts removed from these plants were subject to follow-on mechanical testing and hot cell examination. This follow-on testing confirmed the NDE results.

The program also included inspection of two plants with solution-annealed (SA) Type 347 stainless steel baffle-former bolting. In one plant, all 728 baffle-former bolts were inspected by UT in 1998, with 55 bolts (7.5%) having indications that exceeded the UT acceptance criteria. At another unit, 639 out of the 728 SA Type 347 stainless steel baffle-former bolts were examined in 1999, with 59 bolts (9.2%) having indications failing to meet the UT acceptance criteria. At the first unit, on-site underwater mechanical testing of the removed baffle-former bolts indicated that the actual number of defective bolts was lower than suggested by the UT inspection. However, these known European or domestic baffle-former bolt IASCC indications are not necessarily applicable to all PWR designs. To date, the incidents have been generally associated with CW Type 316 stainless steel or SA Type 347 stainless steel.

Bolts fabricated from SA Type 304 stainless steel appear to be less susceptible. An inspection was performed at one B&W-designed unit in 2005 on all 864 baffle-former bolts and no UT indications were observed.

In 2010, one Westinghouse plant reported finding several broken SA Type 347 stainless steel baffle-former bolt heads and Type 304 stainless steel locking bars on the lower core plate during a normal refueling outage. Subsequent investigation identified a region containing approximately 40 broken or severely damaged bolts. The damage was limited to the upper area of a single large baffle plate.

Visual and volumetric inspections of baffle-former bolts have been conducted under the guidance of MRP-227 at a number of US B&W and Westinghouse-design units. Three B&W units performed baffle-former bolt inspections between 2012 and 2014. All of the bolts were original and constructed of SA Type 304 SS. Of the 2,586 bolts inspected, only 1 displayed crack-like indications. Six bolts were not inspectable due to fit-up problems between the UT probe and bolt heads.

Nine Westinghouse units performed baffle-former bolt inspections between 2011 and 2014. The majority of the bolts were original and constructed of SA Type 347 SS; however two of the units also inspected a number of replacement CW Type 316 SS bolts, previously installed under the baffle-former bolt program. The bolt designs included both internal and external hex heads. The internal hex heads did present some difficulties for proper mating of the UT probe with 58 bolts at one of the units. Of the 6,053 original SA Type 347 SS bolts inspected, 134 or approximately 2% exhibited crack-like indications. None of the 231 replacement CW Type 316 SS bolts inspected exhibited crack-like indications. One unit also reported missing lock bar welds associated with one baffle-former and one baffle-edge bolts; both were assumed to be original fabrication defects.

Starting in 2016, several Westinghouse 4-loop design plants operating in a downflow configuration with Type 347 stainless steel baffle-former bolts observed higher-than-expected levels of baffle-former bolt degradation, including significant clusters of degraded bolts. During the spring 2016 refueling outage, one Westinghouse-designed plant performed MRP-227-A baffle-former bolt inspections. A visual inspection identified 31 baffle-former bolts with protruding heads and lock bar damage, including 2 bolts with missing heads, while the UT inspection identified a total of 182 baffle-former bolts with indications and 14 non-testable bolts, for a total of 227 bolts with potential indications out of a total of 832 bolts. Clustering of the degraded bolts was noted. A visual inspection of the baffle-to-baffle edge bolts and the baffle plates was performed and did not identify any damaged edge bolts or any abnormalities with the baffle plates. Additionally, there were no leaking fuel rods in the core.

Also during the spring 2016 refueling outage, a second Westinghouse-designed plant performed baffle-former bolt visual and UT inspections. During the visual inspection, 18 baffle-former bolts were identified with missing bolt heads or protruding heads. The UT inspection identified 138 baffle-former bolts with indications and 18 non-testable bolts. Additionally, 8 baffle-former bolts were not UT inspected because they were identified to have visually cracked lock bar welds. This represents a total of 182 bolts with potential indications, out of a total of 832 bolts. An additional 7 failures were discovered while replacement work was being performed, bringing the total to 189. Clustering of the degraded bolts was noted. A visual inspection of the baffle-to-baffle edge bolts and baffle plates was performed and did not identify any damaged edge bolts or abnormalities in the baffle plates. There was one leaking fuel rod in the core. The leak mechanism was debris fretting from a separated bolt head or lock bar and not from baffle jetting.

In 2017, multiple Westinghouse-designed downflow plants with Type 347 stainless steel baffle-former bolts performed MRP-227-A baffle-former bolt inspections. One plant identified 256 bolts with indications during a UT inspection of 829 bolts. A later UT inspection was performed during the spring 2019 refueling outage, observing 12 additional bolts with indications out of 562 bolts inspected. Another plant identified 48 bolts with indications during a UT inspection of 828 bolts during its 2017 inspection. A later UT inspection was performed during the spring 2019 refueling outage, observing 13 additional bolts

with indications out of 620 bolts inspected. Out of the remaining plants that performed baffle-former bolt inspections in 2017, one identified 9 bolt indications during a UT inspection of 832 bolts, another identified 1 bolt indication during a UT inspection of 832 bolts, and another identified 0 bolt indications during a visual inspection of 832 bolts.

In the spring refueling outage of 2018, three other Westinghouse-designed downflow plants with Type 347 stainless steel baffle-former bolts performed MRP-227-A baffle-former bolt inspections. One plant identified 3 bolts with indications during a UT inspection of 831 bolts, another plant identified 13 bolts with indications during a UT inspection of 554 bolts, and the third plant identified 6 bolts with indications during a UT inspection of 825 bolts.

Little IASCC has been observed aside from the baffle-former bolt experience. One potential area with experience is the core barrel. During spring 2018 inspections, one CE-designed plant identified crack-like surface indications at the core support barrel (CSB) assembly welds, specifically 1 vertically oriented indication at the middle girth weld and 45 indications adjacent to the middle axial weld. The majority of the middle axial weld indications were oriented perpendicular to the weld, circumferential to the barrel. A supplemental volumetric examination to characterize the indications was subsequently performed. The examination confirmed that the visually identified indications did not extend through the CSB thickness. These flaws were located in material with neutron dose levels high enough for potential IASCC, but the degradation mechanism has not been confirmed. The potential for cracking of CE-designed CSBs and Westinghouse-designed core barrels (CBs) was acknowledged in the previous I&E guidelines; however, the observed condition was inconsistent with expectations for number, location, and orientation of the indications. The OE was communicated to the PWR fleet in MRP 2018-028 [44].

Flow-induced Vibration – In the earlier PWRs, a number of incidents occurred indicating that thermal shields and their support system could be vulnerable to the high flow forces in the vessel-core barrel downcomer. Westinghouse, CE, and B&W responded to these experiences in different ways. The Westinghouse approach was to add vibration-resistance to the shields and to embark on a program to develop advanced thermal shield designs for future plants. For CE plants, thermal shields were removed from operation for all but one facility, which has maintained integrity through positioning pin replacement, tightening, and inspection. The B&W approach was to modify and repair the thermal shields for improved resistance to vibration.

The dominant degradation mechanisms in thermal shields are high-cycle fatigue and SCC resulting from flow-induced vibration, with mechanical wear as a potential consequence. These degradation events appeared predominantly in the earliest thermal shield designs. Typically, the degraded components were fasteners or thermal shield support structures, not the thermal shield itself.

Two CE plants reported cases where failures in the thermal shield resulted in damage to the core barrel. The thermal shields were removed from both plants and the damage to the core barrels was mitigated. One of these two plants with past thermal shield experience is the same plant mentioned in the IASCC section above with regard to potential IASCC-related degradation of the core support barrel.

Three early Westinghouse plants identified thermal shield degradation. The thermal shield degradation in these three plants was repaired; however, they are no longer operating and no operating plant has the same thermal shield design. Two additional Westinghouse plants have reported isolated failures of core barrel bolting that may be linked to flow-induced vibration.

Loss of Material

Wear – Wear of the in-core instrumentation thimble tubes was observed in the top part of the Zircaloy-4 thimble tubes at three CE-designed units. These tubes experienced through-wall tube degradation as a result of flow-induced vibration in the vicinity of the fuel alignment plate. This particular wear phenomenon was addressed by making modifications to the fuel alignment plate to alter the flow conditions in the vicinity of the entry point of the thimble tubes into the plate. Wear as a result of flow-induced vibration has not been observed in these components after implementing the modifications to the fuel alignment plate. Accordingly, these components are not considered susceptible to this type of wear in the future.

Problems were noted involving the original locking devices for the B&W-design vent valve jackscrews in the late 1970s and early 1980s. The jackscrew locking mechanism was vibrating and wearing through the locking cup. A new locking mechanism was designed and supplied to most B&W units. At least four of the eight vent valves were modified with the redesigned locking devices. The four vent valves next to the two outlet nozzles were replaced. In the late 1970s and early 1980s, problems were also noted involving the original jackscrew guide bushing, which was found to be improperly secured on some valves. Procedures were developed to install the modified locking device on the jackscrew and to secure the bushing when necessary.

Wear of the Westinghouse control rod guide tube assembly guide cards has been reported at several domestic and international plants. The wear enlarges the guide card holes that guide the control rods through the assembly and maintain the alignment of the rods. Specific guidelines for the visual inspection and evaluation of guide card wear have been established by the PWROG (ref. EPRI letter MRP-2014-006) and incorporated into Rev. 1 of MRP-227. The PWROG guidelines utilize operational time extension curves to determine the remaining life of guide cards. These curves are also utilized in conjunction with empirical wear data to establish the scope and timing of initial and subsequent inspections.

Six U.S. Westinghouse design units performed inspections of guide cards within 124 guide tube assemblies under the guidance of MRP-227 and WCAP-17451-P between 2012 and 2014. Significant wear of isolated guide cards within two guide tube assemblies was observed at two, 14x14 units; however the associated guide tube assemblies were deemed suitable for continued service. The remaining units showed only minor guide card wear

In 2016, based on OE from a European utility regarding aggressive guide card wear for 17x17 AXLR guide tubes, several foreign material exclusion (FME) videos of Westinghouse-designed plants with 17x17 A/AS/AXLR guide tubes and ion nitride RCCAs were assessed for guide card wear. Although 17x17 A/AS/AXLR guide tube FME videos were reviewed in WCAP-17451-P [37], none of these plants had ion nitride RCCAs. The evaluations indicated that the guide tube guide cards were wearing at an accelerated pace for this combination of guide tube and RCCA. The wear did not appear to be significant enough to create a nuclear safety concern, but guide card wear measurements were taken during the fall 2016 outage at one of the plants in order to better quantify the wear. The guide card wear measurement data confirmed the presence of accelerated guide card wear with 17x17 A guide tubes and ion nitride RCCAs. Further evaluation of the guide card wear measurement data indicated that one guide tube needed to be taken out of service and that several others would need to be replaced soon after. The guide tube taken out of service was judged to have less than one remaining functional cycle of operation. Three more plants with this combination of guide tube and RCCA have been inspected since. Two of these three plants have aggressive guide card wear, and the other plant has wear that could be categorized as moderate. One additional difference in the four plants inspected in the U.S. with 17x17 A/AS guide tubes and ion nitride RCCAs is that the plant with moderate wear uses a heavier absorber rodlet, categorized as Ag-In-Cd, versus the plants with aggressive wear, which use B_4C pellet absorber material.

A total of approximately 28 plants have performed guide card wear measurements in the U.S. to date. The breakdown of these 28 plants with respect to guide tube design is as follows:

- 14x14 – 4 plants
- 15x15 – 7 plants
- 17x17 Std. – 10 plants
- 17x17 A/AS – 7 plants

The wear at 14x14 and 15x15 plants is mostly categorized as low. Moderate to high (aggressive) wear was noted at 17x17 guide tube plants.

The wear surfaces on the radial keys and clevis inserts are routinely examined as part of the PWR internals ASME B&PV Code Section XI inservice inspection programs. While reports of scratches, superficial wear, or both are common in these inspections, one European plant has reported significant wear scars at these surfaces. Guidance for inspection of radial key and clevis insert interfacing surfaces has been provided by Westinghouse in TB-14-5.

In all currently operating Westinghouse and B&W plants, the incore flux detectors are directed through the RV bottom head via thimble tubes or guideways. For the bottom-mounted instrumentation design, the thimble tubes are retractable, and the insertion and retraction of these tubes are directed by long-radius guides below the bottom head and by internals guides between the bottom head and fuel assemblies. There is significant variation among plants with regard to thimble tube diameters (outer and inner), thimble tube-to guide path clearance, length of thimble tube exposed to coolant, and flow conditions. No wear concerns have been identified in the B&W design.

The primary historical concerns with flux thimble degradation in Westinghouse-designed plants have been obstruction of the flux detector pathways, wear due to flow-induced vibration of the thimble tube, flow-induced vibration fatigue damage to thimble tube guideways, and damage to in-core instrumentation flange seating surfaces at refueling. The obstruction problem can often be mitigated by appropriate cleaning procedures at refueling. All Westinghouse plants are required by NRC Bulletin 88-09 to have an inspection program to periodically confirm incore neutron monitoring system thimble tube integrity. Reductions in wall thickness due to wear are normally monitored with an eddy-current inspection. Many plants have chosen to replace the flux thimbles with improved designs. These programs have been successful in managing thimble tube degradation.

A visual inspection in 1973 at one CE-designed plant revealed worn areas in the RV flange and head resulting from inadequate hold-down spring design and subsequent PWR internals vibration. Prior to shutdown, higher than normal ex-core neutron detector readings had suggested the possibility of excessive internals vibration. Wear was found on the mating surfaces, alignment keys and slots, snubbers, and outlet nozzle faces. The worn surfaces were repaired and a new design using Belleville spring assemblies greatly increased hold-down capacity and mitigated the issue.

Three Westinghouse design units performed measurements of 304 SS hold-down spring height between 2012 and 2014. All three springs displayed acceptable levels of relaxation that indicated adequate hold-down capability would be maintained for at least 60 years of operation.

Change in Dimension

Irradiation-Induced Growth – Although irradiation-induced growth of zirconium alloys in CE-designed plants was not explicitly identified in MRP-175 as an age-related degradation mechanism to be evaluated as part of the screening process, irradiation-induced growth in the axial direction of the in-core instrumentation thimble tubes has reduced the clearance between the thimble nose and the bottom of the fuel assembly. Some plants had observed that the thimble tube support plate was raised above its normal support position when the upper internals structure was set in place after fuel reload. This indicated that for some of the thimbles the gap tolerance between the thimble tube and the bottom end fitting of the fuel assembly had been reduced until the tube contacted the bottom fitting of the fuel assembly and was being loaded in compression. Ten plants affected by this issue have taken actions. Six of these plants have already replaced the thimble tube assemblies with modified designs that are shorter in length to accommodate the expected irradiation-induced growth. Two additional plants have replacement designs in fabrication and have made preparations to install the replacement thimbles in an upcoming outage. The remaining two plants have not yet begun preparations for a full replacement of the thimble tubes, but one of these two has instead taken the intermediate step of raising the thimble support plate to accommodate additional axial growth. These plants are planning to execute a thimble assembly replacement program during a future refueling outage that is not currently encumbered with other large-scale replacements of major components. All affected plants will likely have replaced their thimble tubes prior to license extension.

Miscellaneous

Barrel-Former Bolting **–** In the mid-1990s, one Westinghouse plant reported finding a type 347 stainless steel barrel-former bolt on the lower core plate during a refueling outage. Subsequent investigation identified a total of three baffle-former bolts in the top former row which were either loose or completely displaced. The bolts and associated holes all had observable thread wear. All bolt heads were completely or partially intact; there was not complete separation at the head-to-shank interface on any of the three specimens. Stress corrosion cracking was ruled out as a factor in the mode of failure. Contributing causes were reported as elevated bolt stress near the thermal shield support block and bending stress on the bolts during normal steady state operation, but no specific root cause was identified.

B&W-design impact damage **–** Vent valve damage has been observed at some units, which was due to an interaction with the plenum assembly during insertion and removal activities. Items include jackscrew locking cups, vent valve bodies, and guide blocks. One vent valve was identified to have the key ring not engaged on both jackscrews. Vent valves are replaceable items and as noted above, have been replaced as necessary. In addition, one B&W unit identified a guide block bent outward from the lower grid shell forging. Marks on the guide block and core barrel stand leg indicate the damage may have occurred during a previous outage when setting the core barrel in the stand.

Mechanism Unidentified to Date – Visual examinations at one B&W-designed unit in 2005 indicated that three or four internal baffle-to-baffle bolts were found protruding. The bolt heads extended beyond the baffle plate surface. This was an indication that the locking devices, and potentially the bolts as well, had failed. As noted above, a UT inspection of 100% of the baffle-former bolts was performed, with no detected indications of broken bolts. No UT inspection was performed on the internal baffle-to-baffle bolts, and the potentially failed baffle-to-baffle bolts have yet to be removed to confirm failure and, if failed, the mechanism. As a result of the observations, AREVA (now Framatome) performed a unit-specific evaluation to assess operational and safety functions for continued operation. That evaluation included thermal hydraulic evaluation, structural evaluation, fuel evaluation, and loose parts evaluation.

In 2014, one B&W unit inspecting the IMI guide tube spiders and spider-to-lower grid rib section welds identified two linear indications. One indication was located at the top of the CASS spider next to the weld, the other indication was located in the Type 304 lower grid rib section below the weld.

Reactor Internals Component Replacements – Replacement of upper internals in Westinghouse and CE designs have been made.

Beginning in 2004, replacement of the complete internals (upper and lower internals) at three Japanese PWRs has also been performed. It has been stated that these replacements have been performed for the following reasons:

1. To keep and improve operational reliability, safety, and a high load factor for the nuclear power units
2. To maintain the plant against aging degradation of the PWR internals
3. To mitigate degradation risks that would rise with increasing operational time in the future

Appendix B: Guidance on Plant-Specific Applicability Related to Fuel Design or Fuel Management

MRP-227-A Applicant Licensee Action Item 1: MRP-227-A Applicability Guidelines for Combustion Engineering and Westinghouse Pressurized Water Reactor Designs (*Ref. WCAP-17780-P, MRP-2013-025, and ML14309A484*)

Several public meetings were held in 2012 and 2013 with the NRC staff to discuss NRC expectations and concerns regarding industry responses to MRP-227-A, A/LAIs 1 and 2. The concerns were addressed to owners of currently operating pressurized water reactor plants designed by Westinghouse and CE. At these meetings, the NRC, Westinghouse, EPRI, and utility representatives discussed regulatory concerns and determined a path for a comprehensive and consistent utility response to demonstrate applicability of MRP-227.

The information provided by the industry to the NRC staff demonstrated that the MRP-227 I&E Guidelines are applicable for the range of conditions expected at the currently operating Westinghouse and CE-designed plants in the U.S. As a result of the technical discussions with the NRC staff, this Appendix was developed to provide utilities with the basis for a plant to respond to the NRC's Request for Additional Information (RAI) to demonstrate compliance with the basic technical applicability assumptions in MRP-227 for originally licensed and uprated conditions.

BACKGROUND

The Safety Evaluation (SE) issued on MRP technical report MRP-227, Revision 0, by the U.S. NRC contained eight A/LAIs. These eight action items must be completed in the implementation of the I&E Guidelines outlined in MRP-227-A.

On November 28, 2012, a public meeting was held at the NRC office to discuss staff expectations and concerns regarding industry responses to A/LAIs 1 and 2. The concerns were addressed to owners of currently operating pressurized water reactor plants designed by Westinghouse and CE. A series of proprietary and public meetings were conducted from January to June of 2013. At these meetings, the NRC, Westinghouse, EPRI, and utility representatives discussed regulatory concerns and determined a path for a comprehensive and consistent utility response to demonstrate applicability of MRP-227.

Westinghouse summarized the proprietary meeting presentations and supporting proprietary generic design basis information in WCAP-17780-P, and provided it to the NRC. WCAP-17780-P provides background proprietary design information regarding variances in stress, fluence, and temperature in the plants designed by Westinghouse and CE to support NRC reviews of utility submittals to demonstrate plant-specific applicability of MRP-227. NRC staff assessed this information provided in WCAP-17780-P and EPRI MRP-2013-025 as documented in ML14309A484 [43].

Plant-specific evaluation to demonstrate the applicability of MRP-227 for managing aging would need to consider the following items:

1. designated design specific criteria in responding to specific NRC requests for additional information,
2. criteria defined in MRP-227, Section 2.4, and
3. plant-specific regulatory commitments for managing aging in PWR internals.

The NRC staff subsequently stated that the information provided by the industry to the NRC staff demonstrated that the MRP-227 I&E Guidelines are applicable for the range of conditions expected at the currently operating Westinghouse and CE-designed plants in the U.S.

As a result of the technical discussions with the NRC staff, the basis for a plant to respond to the NRC's Request for Additional Information (RAI) to demonstrate compliance with MRP-227 for originally licensed and uprated conditions was determined to be satisfied with plant-specific responses to the following question related to plant-specific Fuel Design and/or Fuel Management:

Question: *Does the plant have atypical fuel design or fuel management that could render the assumptions of MRP-227, regarding core loading/core design, non-representative for that plant?*

GUIDANCE ON PLANT-SPECIFIC APPLICABILITY DEMONSTRATION

Guidance for demonstrating compliance with MRP-227 conditions in responding to NRC questions and assessments for uprated conditions, if applicable, is provided in the following paragraphs.

Fuel Design or Fuel Management - Question Response Guidance:

The MRP-227 inspection recommendations require extensive inspections of the components surrounding an active core. Additional inspection requirements are not anticipated. The aging degradation analysis completed in the development of MRP-227 indicated decreased degradation rates under the low leakage core loading assumptions assumed during the second 30 years of operation. To provide assurance that there would not be higher than anticipated rates of degradation in the later years of operation, MRP-227 guidance on applicability of the recommendations precludes return to out-in core loading patterns. MRP-227 does not provide a quantitative definition of out-in core loading that could be used to evaluate the potential impact on internals degradation. The primary impact of out-in core loading is increased rates of nuclear heat generation in the PWR internals.

Plant-specific fuel design or fuel management application were identified as having potential to invalidate the fluence, temperature, and stress assumptions used to develop MRP-227. The goal in the development of this guideline was to define a simple parameter to demonstrate that the assumptions of MRP-227 are representative of the plant. Subsection 4.3.2 of MRP-191 states that:

"*The core power density that was assumed for projecting the fluence values for the Westinghouse plant was 104.5 W/cm^3. The core power density assumed in the analysis of the CE-designed plant was*

83.0 W/cm^3. Higher core power densities result in higher neutron fluence values."

The largest impact of changes in the fluence distribution on recommendations occurs in components near the lower bound fluence for susceptibility. Any limit on fluence or on the heat generation rate is effectively a limit on core power spatial distribution.

Three different boundaries were explored in developing this guideline:

1. radial boundary (components laterally surrounding core)
2. upper axial boundary (components above core)
3. lower axial boundary (components below core)

Radial Boundary Limitations

The primary driver for the radial core power distribution is the MRP-227 basis of 30 years of "out-in" management, where fresh fuel is placed in peripheral core locations, followed by 30 years of low leakage fuel management. Any change in this scenario has the potential to impact reinspection, but not the initial inspection timing or affected components. Due to design similarities across the currently operating Westinghouse and CE U.S. fleet, in most, but not all cases, internals component geometry is a secondary effect. Neutron flux and heating rate could be expected to vary by as much as a factor of 5, depending on radial core power distribution and absolute rated power.

Local effects at key locations are dominated by a few (typically 3-5) fuel assemblies located on the core periphery. There is no impact on the initial inspection, but a change from the low leakage operating characteristics during the second 30 years of operation could impact the MRP-227 reinspection recommendations. To provide assurance that there would not be higher than anticipated rates of degradation in the later years of operation, MRP-227 guidance on applicability of the recommendations precludes return to out-in core loading patterns. The limitations on power for demonstrating applicability in the peripheral assemblies provided in this guideline preclude return to the more damaging out-in core loading pattern.

Plant-specific applicability of MRP-227 in the radial direction with no further evaluation required is demonstrated by meeting the following limits:

<u>CE</u>: heat generation figure of merit, $F \leq 68$ Watts/cm^3
average core power density < 110 Watts/cm^3

<u>Westinghouse</u>: heat generation figure of merit, $F \leq 68$ Watts/cm^3
average core power density < 124 Watts/cm^3

The figure of merit (refer to MRP-2013-025 Figure 1), F, is given by:

$$F = Pavg\ (W1{*}R1 + W2{*}R2 + W3{*}R3 + W4{*}R4)$$

Where:

Pavg = average core power density
Wi = generic inside corner weighting factor (MRP-2013-025 Figures 1 and 2)
Ri = relative fuel assembly power

The limiting core power values determined for the plant-specific assessment apply to operation beyond the first 30 years. Plants that do not meet these guidelines may require

additional evaluation to demonstrate the applicability of the MRP-227-A recommendations. Plants that exceed the radial boundary limitations should evaluate the impact on the reinspection schedules for components in the Westinghouse baffle/former/barrel/shield or the CE core shroud/barrel/shield structure.

Upper Axial Boundary Limitations

The limitations for the upper axial components or components above the reactor core, typically the Westinghouse upper core plate or the CE fuel alignment plant, are based on the MRP-175 and MRP-191 fluence threshold for irradiation embrittlement. The primary driver is the fuel assembly geometry, noting the position of the active fuel stack and the use of axial blankets.

Similar to the radial components due to design similarities across the fleet, in most cases, upper axial component geometry is a secondary effect. Contrary to the radial boundary components, out-in versus low leakage operation has only a small effect on the maximum exposure. Neutron flux and heating rate above the reactor core could be expected to vary in the fleet by as much as a factor of 4, depending on fuel assembly design, core power distribution, and absolute rated power.

Variations could impact screening results for components above the core; some plants may exceed the screening criterion, others may not.

Plant-specific applicability of MRP-227 in the upper axial direction with no further evaluation required is demonstrated by meeting the following limits:

CE: active fuel to fuel alignment plate distance > 12.4 inches
average core power density < 110 Watts/cm^3

Westinghouse: active fuel – upper core plate distance > 12.2 inches
average core power density < 124 Watts/cm^3

Evaluations shall consider the entire plant operational period, original and extended life. Plants that exceed the limits for more than a cumulative two years of operation shall perform further evaluations to demonstrate compliance with this applicability requirement. A plant-specific analysis may be required to demonstrate that the fluence above the upper core plate or fuel alignment plate does not exceed the irradiation embrittlement screening threshold. In the event that this fluence limit is not met, an evaluation of potential irradiation embrittlement in components immediately above the plate may be required.

Lower Axial Boundary Limitations

The limits for the lower axial components or components below the reactor core were evaluated based on the MRP-175 and MRP-191 fluence threshold for irradiation embrittlement.

The parameters affecting the lower axial components are identical to those for the upper axial components. The primary driver is the fuel assembly geometry. Due to design similarities

across the fleet, in most cases, lower axial component geometry is a secondary effect. Out-in versus low leakage operation has only a small effect on the maximum exposure. Neutron flux and the heating rate below the reactor core could be expected to vary in the fleet by as much as a factor of 4, depending on fuel assembly design, core power distribution, and absolute rated power.

However, the variations do not impact the MRP-227 recommendations for managing aging for the lower axial components in the currently operating CE and Westinghouse fleet.

Plant-specific applicability of MRP-227 in the lower axial direction with no further evaluation required is demonstrated by meeting the MRP-227, Section 2.4 criteria.

Extended Power Uprate (EPU)

Additionally, further consideration related to Extended Power Uprates need to be addressed, specifically:

> "*If the plant implemented an Extended Power Uprate (EPU), are the peak internal metal temperatures within the assumptions made in developing MRP-227?*"

If the plant implemented an EPU, all changes should be assessed against the plant-specific AMR, aging management plan, and any other pertinent documents that support the aging management of PWR internals (including physical modifications to PWR internals supporting the EPU). The above question related to fuel design or fuel management should be evaluated for the EPU, and responses should be provided as applicable.

If there were no physical modifications to the PWR internals as a result of the EPU, then the applicability requirements outlined above should be re-examined for EPU conditions. An EPU results in increases in average core power. The MRP-227 I&E Guidelines should remain applicable as long as the above discussed average core power and peripheral heat generation limits imposed remain bounding. If these criteria are not met, plant-specific evaluations may be required to demonstrate the applicability of the guideline.

Thus, the utility response with regard to the applicability of MRP-227 for the EPU uprated condition should consist of appropriate text to identify the uprated plant condition and comparisons with the above criteria.

SUMMARY

To demonstrate plant-specific applicability of the MRP-227 sampling inspection strategy for managing aging in PWR internals, licensees must demonstrate that the criteria of MRP-227, Section 2.4 are met, and that the neutron fluence and heat generation rates are within the range of the following variables summarized. The limiting threshold values are:

- active fuel – upper core plate distance > 12.2 inches for Westinghouse plants
- active fuel to fuel alignment plate distance > 12.4 inches for CE plants
- average core power density < 124 Watts/cm^3 for Westinghouse plants
- average core power density < 110 Watts/cm^3 for CE plants
- heat generation figure of merit, F ≤ 68 Watts/cm^3 for Westinghouse and CE

Plants that exceed the thresholds defined above do not necessarily fall outside the MRP-227 recommendations; instead, they require additional evaluations to fully demonstrate plant-specific applicability. The limiting core power values apply to operation beyond the first 30 years, and should be based on the full as-licensed operational (original and extended life) life of the plant.

A plant that maintains core loading patterns that meet the limits would satisfy the MRP-227 requirement to avoid operation above this limit. Short periods of operation (fewer than two years total) above this limit would not invalidate the requirement to not return to "out-in" fuel management.

Plants that exceed the above limits for more than a cumulative two years of operation would need to provide further evaluations to demonstrate compliance with this applicability requirement.

Appendix C: Summary of Changes to Document

Part 1 - Summary Description of Document Changes from MRP-227-A Revision 0 to MRP-227 Revision 1

Page[(1)]	Section[(1)]	Description of Change
1-1	1	Added clarification to Executive Summary associated with ASME Boiler and Pressure Vessel Code Section XI, and crediting aging management for 'existing programs'.
1-2	1	Added statement at end of Executive Summary regarding the comprehensive acceptance methodology / guidance is detailed in the technical report WCAP-17096. (Ref.[27])
2-1	2.1	Added clarification to the Background section regarding the definitions of 'engineering program' and 'aging management program' as well as specific relevant information about the purpose(s) of the Revision 1 changes.
2-4	Figure 2-2	Clarified the "Analysis" section of this figure to state "Engineering Evaluation+Assessment" instead of "Functionality Assessment" under the "Category C" section.
2-5	2.3	Added statement regarding non-applicability of I&E guidelines to plants beginning construction after 2007.
2-5	2.4	Added clarifying statements to Section 2.4 "Guidelines Applicability" regarding expectations for users of these I&E guidelines, as well as general assumptions associated with the applicability of these guidelines. Also, included additional guidance information regarding applicability of guidelines based on core power characteristics as detailed in new Appendix B which is the technical information from MRP-2013-025 and WCAP-17078-P. Additionally, included guidance information related to evaluating ASME Section XI 'existing programs' for internals, and use of FMECA process(es) and assessments of reactor internals fabricated from CASS.
3-5	Figure 3-2	Replaced original figure with better representation.
3-5	3.1.2	Added clarifying information related to CE design core barrel welds and design-related characteristics.
3-9	Figure 3-5	Replaced original figure with better representation.
3-11 and 3-13	3.1.3 and 3.2.6	Added reference call out [27] related to MRP-276 technical report.
3-15	3.3.1	Added 'e.g.' call out related to ASME B&PV Code Section XI programs as 'existing programs'. Also, clarified text detailing that the MRP-227 I&E guidelines are not intended to supersede the ASME code at bottom of page.
3-16	3.3.2	Deleted the three (3) bullets from end of section 3.3.2. Included the information from these three (3) bullets related to design-specific information in Section 3 Tables related to Existing programs components to table footnotes, as appropriate for each NSSS design-type.
3-20	Table 3-1	Clarified footnotes as appropriate (above) and added new footnotes based on plant-specific reviews.
3-25	Table 3-3	Split row associated with Core Barrel Girth Weld and added clarifications related to location of specific welds
3-26	Table 3-3	Deleted specific information related to Final Disposition/Categorization of Westinghouse clevis insert bolts (OE) for wear. (Ref. Existing Westinghouse Technical Bulletin TB-14-5.)

Page[(1)]	Section[(1)]	Description of Change
4-1	Sect. 4	Added clarification statement regarding "through application of these guidelines" at end of first paragraph.
4-1	Sect. 4	Added 'elsewhere' at end of last paragraph.
4-2	Sect. 4	Added "e.g." to parenthetic statement on top of page.
4-2	Sect. 4	Clarified last three (3) paragraphs of opening portion of Section 4 regarding 'intent of these guidelines'.
4-3	Sect. 4.1	Re-organized first paragraph into numeric list and clarified some major element descriptions
4-3	Sect. 4.1.1	Added "including overall sampling strategy"
4-3 + 4-4	Sect. 4.1.3	Added "MRP-228" in two places.
4-5	Sect. 4.2.1	Added clarifying information regarding use of visual VT-3 exams in industry, specific requirements for VT-3 resolution requirements, as well as additional information related to detection of IGSCC cracking in GE-design BWR plant core shrouds. Deleted last paragraph in section 4.2.1 related to character height requirements for VT-3.
4-5 + 4-6	Sect. 4.2.2	-Clarified wording associated with use and capability of VT-1 and EV-1 inspection methods. -Added last sentence clarifying specific details related to ASME B&PV code character discrimination. Also, added "MRP-228" in one location.
4-7	New 4.2.6	Added new Section 4.2.6 regarding "Subsequent Examination Intervals"
4-7	New 4.2.7	Added new Section 4.2.7 regarding "Coordination with ASME Section XI Code Requirements"
4-8	Sect. 4.3	Added additional paragraph discussing "general guidance for selection and justifying condition monitoring inspection populations and sample sizes" per NUREG-1800 Rev.2
4-9	Sect. 4.3	Added final paragraph discussing "continued implementation of existing guidance" for Existing Programs components is consistent with this aging management approach
4-9- 4-12	Sect. 4.3.1	Deleted descriptions of Primary and Expansion linkages for B&W plants; see tables
4-14- 4-16	Sect. 4.3.2	Deleted descriptions of Primary and Expansion linkages for CE plants; see tables
4-17- 4-19	Sect. 4.3.3	Deleted descriptions of Primary and Expansion linkages of Westinghouse plants
4-20- 4-45	Figure #s	Renumbered various figures and figure call-outs in Tables as necessary.
4-20- 4-25	Table 4-1 and notes	Several clarifications to Primary components items as well as a few new additions of primary components based on plant-specific reviews completed for B&W plants.
4-26- 4-31	Table 4-2	-Clarifications to naming of Core Support Barrel Assembly welds, as well as changes to Examination Coverage requirements for these welds to 25% sampling of weld circumference. -Adjustments as necessary for primary component items of CE plants (e.g., core barrel welds). -Editorial adjustments to note call-outs and Notes at the end of table
4-27	Table 4-2	Changes in Examination coverage for CE Welded Core Shroud Assembly in CE plant designs with core shrouds assembled in two vertical sections, relocated supplementary information regarding gap measurements to Note 4 at end of table

Page[(1)]	Section[(1)]	Description of Change
4-32	Table 4-3	-Changes in primary component Control Rod Guide Tube Assembly Guide Plates (Cards) examination requirements to be consistent with MRP-2014-006 and WCAP-17451-P. -Added new Note 7.
4-32- 4-35	Table 4-3	-Clarifications to naming of Core Barrel Assembly welds, as well as changes to Examination Coverage requirements for these welds to 25% sampling of weld circumference. -Adjustments as necessary for other primary component items of Westinghouse plants (e.g., core barrel welds). Editorial adjustments to note call-outs and Notes at the end of table
4-35	Table 4-3 notes	Specific (as well as editorial) changes to notes, and added new notes 7-10 as necessary.
4-36- 4-39	Table 4-4 and notes	Editorial clarifications and additions to Expansion Component items for B&W plants, based on plant-specific reviews performed to-date.
4-40- 4-42	Table 4-5 and notes	-Added new Expansion row for clarification of CE Plant Core Shroud Assembly (Welded) Remaining Axial Welds for plant designs with core shrouds assembled in two vertical sections. -Editorial clarifications to note call-outs and specific wording in notes at end of Table 4.5 for CE plants. Deleted previous Note 2 regarding minimum of 75% coverage sample.
4-43	Table 4-6	Changed examination method for Westinghouse expansion components Upper Core Plate and Lower Support Forging or Castings from EVT-1 to VT-3.
4-43- 4-45	Table 4-6 and notes	Changes to Primary to Expansion links as necessary for Westinghouse Expansion Components as well as changes to naming of and inspection requirements for Core Barrel girth welds and axial welds as detailed in descriptions in Section 4.3.3. Deleted row associated with Core Barrel Outlet Nozzle welds. Edited Note numbers as necessary.
4-44	Table 4-6	Combined rows associated with Westinghouse Expansion Components Lower Support Assembly Lower support column bodies (Cast and Non-Cast) into single row, with clarified examination method for VT-3 and examination coverage of 25% of column assemblies as visible from above the lower core plate.
4-45	Table 4-6	Added row associated with Westinghouse Expansion Component Control Rod Guide Tube Assembly Remaining CRGT lower flange welds, with linkage to Primary component from CRGT Lower Flange Welds.
4-46	Sect. 4.4	Added discussion and edited section 4.4 to clarify the Existing Programs Component Requirements as credited per ASME Section XI Table IWB-2500 and other ancillary designations, for meeting the aging management requirements and these I&E guidelines. These clarifications expound upon the consideration that Existing Programs are "credited within these guidelines as adequate for continued aging management during the extended period of operation."
4-47	New Sect. 4.5	Added new Section 4.5 regarding "No Additional Measures Components" for clarification of considerations of components that are subject to in-service inspection requirements per ASME Section XI IWB-2500, as well as reminder regarding need for utility relief requests.
4-48	Table 4-8 and Notes	Added row associated with CE plant design Existing Requirement for Alignment and Interfacing Component Core Stabilizing Lugs and Shims, per ASME Code Section XI. Added new notes as necessary.

Page[1]	Section[1]	Description of Change
4-50	Table 4-9 and Notes	Added reference to "IEB-88-09" in row associated with BMI Flux Thimble Tubes item. Deleted Note 2.
4-52- 4-89	Section 4 figures	-Replaced previous Figure 4.9 with new Figure 4.9 regarding B&W Vent Valve Body outside-view, and Figure 4.10 regarding BB&W Vent Valve Body inside-view. Previous Figure 4.9 showed B&W CSS Outlet Nozzle which is not applicable for reactor internals program. -Replaced many Sect. 4 figures with clearer and more meaningful figures, and also added some new figures developed recently -Relocated figures into one section of pages near end of Section 4.
5-2- 5-5	Table 5.1 and Notes	-Clarifications and editorial changes to Table 5.1 regarding B&W-design examination criteria. -Some new Primary items added based on changes in Section 4 tables and from various plant-specific reviews performed to date. -Added specific information to first row of Table 5.1 regarding Plenum Cover Support Ring -Added specific information to second row of Table 5.1 regarding VT-3 of CRGT spacer castings and expansion to vent valves -Added new item to Table 5.1 specific to TMI-1 for Shock pad bolts and locking devices -Added new rows to Table 5.1 related to Vent Valve Assemblies for original and modified type -Added specific information to last row of Table 5.1 regarding IMI grid rib section welds. -Editorial changes and clarifications to note call-outs and new notes added at end.
5-6- 5-11	Table 5.2	-Added clarifying information related to CE shroud assembly shroud plates. -Renamed Upper CSB flange weld to Upper flange weld (UFW), included modified expansions -Renamed lower cylinder girth welds to Middle girth weld -Included new item to Table 5.1 related to CSB flexure weld -Clarified row for Lower Support Structure Core support columns
5-12- 5-14	Table 5.3 and Notes	-Added clarifying information related to Westinghouse guide plates (cards) per MRP-2014-006 and WCAP-17451-P reference documents. -Clarified expansion link components and criteria for CRGT assembly lower flange welds -Clarified upper flange weld (UFW) and expansion links and criteria -Added new row for Core Barrel Assembly Lower Girth Weld with expansion links/criteria -Clarified row associated with Baffle Former Assembly (includes: baffle plates, etc.) -Clarified Note 2
5-17	5.2	Expanded / clarified first paragraph in Section 5.2 regarding physical measurements
6-2-	6	Deleted majority of Section 6; clarified remaining text to point to WCAP-17096-NP
7-1- 7-2	7	Clarified sections 7.2, 7.3, 7.4 and 7.6 with revised NEI-03-08 implementation requirements. Also, combined original Needed items 7.5 and 7.7 and clarified into one Needed item 7.5
7-2	7.6	Changed inspection results reporting timeframe from 120-days to 6-months.

Page[1]	Section[1]	Description of Change
A-1- A-7	Appx. A	-Added new relevant information into Appendix A for Operating Experience summary related IGSCC, PWSCC, IASCC, and wear mechanisms, as well as some miscellaneous items.
B-1- B-7	Appx. B	-Replaced previous Appx. B (NRC staff SER and RAIs) with new Appendix B regarding Guidance on Plant-Specific Applicability related to Fuel Design or Fuel Management, as per existing reference documents WCAP-17780-P, MRP-2013-025, and ML14309A484.

Part 1 Table Notes:

1. Page numbers and section numbers correspond to MRP-227 Revision 1.

Part 2 - Summary Description of Document Changes from MRP-227 Revision 1 to MRP-227 Revision 1-A

Page(1)	Section(1)	Description of Change
iii	N/A	NRC Safety Evaluation on MRP-227 Revision 1 has been added.
v	N/A	Acknowledgments page has been updated.
vii	N/A	Revision statement added at the end of the first paragraph of the Report Summary. “reactor vessel internal” has been changed to “PWR internals” in the second sentence of the first paragraph. “reactor internals” has been changed to “PWR internals” within the Background paragraph.
vii	N/A	“reactor internals” has been changed to “PWR internals” within the Objectives section and Approach section.
ix and x	N/A	Executive summary of the report has been added.
xi, xii, and xiii	N/A	Added several acronyms and removed other acronyms no longer used.
xv	N/A	Record of revisions has been updated.
1-1	1	“reactor internals” has been changed to “PWR internals” within first paragraph and first bullet point.
1-2	1	“reactor internals” has been changed to “PWR internals” within first paragraph. “Babcock & Wilcox (B&W)” has been updated to “B&W” as the acronym was previously defined. “Combustion Engineering (CE)” has been updated to “CE” as the acronym was previously defined. Added clarification stating, “There are no Existing Programs components for B&W plants.”
2-1	2.1	“reactor internals” has been changed to “PWR internals” within first paragraph. “Pressurized Water Reactor (PWR)” has been updated to “PWR” within the first paragraph as the acronym was previously defined. “*Aging Management Program (AMP)*” was updated to “*Aging Management Program*” as the acronym was previously defined. Minor editorial change in the *Aging Management Program* description.
2-2	2.1	Acronyms added next to degradation effects.
2-5	2.4	“Aging Management Program (AMP)” was updated to “AMP” as the acronym was previously defined.
2-6	2.4	Editorial change from “base loaded” to “base-loaded”.
2-7	2.4	“reactor internals” has been changed to “PWR internals” within last paragraph. Editorial change to last sentence from “provided to utility owners a potentially useful tools,” to “provided to utility owners as potentially useful tools.”
3-6	3.1.2	“in-core instrumentation (ICI)” was updated to “ICI” as the acronym was previously defined.
3-13	3.2	“stress corrosion cracking (SSC)” and “irradiation-assisted stress corrosion cracking (IASCC)” have been changed to “SCC” and “IASCC” as both acronyms were previously defined.
3-14	3.2.1 and 3.2.2	“stress corrosion cracking (SSC)” and “irradiation-assisted stress corrosion cracking (IASCC)” have been changed to “SCC” and “IASCC” as both acronyms were previously defined.
3-14	3.2.5	“cast austenitic stainless steel (CASS)” was updated to “CASS” as the acronym was previously defined.

Page(1)	Section(1)	Description of Change
3-16	3.2.8 and 3.3.1	"irradiation-enhanced stress relaxation (ISR)" was updated to "ISR" as the acronym was previously defined. Editorial change from "were sufficiently unaffected by consequences to be placed into Category A," to "were sufficiently unaffected by consequences and placed into Category A."
3-17	3.3.2	Editorial change in the last paragraph, addition of the NRC SE on MRP-227 Revision 1.
3-19	Table 3-1	Deletion of note 9 for the Upper Grid Fuel Assembly Support Pads: Alloy X-750 Dowel Locking Weld in the Final Group column.
3-24	Table 3-2	Core Support Columns, 304 SS and CF8, were updated from P (Primary) to E (Expansion) and note 5 was added.
3-26	Table 3-2	Table notes were updated to delineate between the NRC SER on MRP-227 Revision 0 and MRP-227 Revision 1.
4-1	4	"aging management program (AMP)" was updated to "AMP" as acronym was previously defined.
4-2	4	Editorial change in fourth paragraph, from "in the Section 4 tables is intended to be the driving; thus," to "in the Section 4 tables is intended to be the driving factor; thus, "reactor internals" has been changed to "PWR internals". "re-inspection" has been changed to "reinspection" within the fifth paragraph.
4-5	4.2.1	Reference added for ML14139A178 at the end of paragraph 4.
4-6	4.2.2	Clarification based on RAI 20 response was added at the end of paragraph 2 stating, "The adjacent base metal to be examined is defined in Section 2.3.6.4 of MRP-228 and is referenced in the Section 4 Primary and Expansion component tables in this document." An acronym for heat-affected zone, (HAZ), was also added to paragraph 2.
4-7	4.2.4	"irradiation-enhanced stress relaxation (ISR)" was updated to "ISR" as the acronym was previously defined.
4-8	4.2.7	"reactor internals" has been changed to "PWR internals".
4-8	4.2.7	"10 year ISI" was changed to "10-year ISI"
4-9	4.3	Comma added in the sentence "Where specified, the technical…"
4-9	4.3	A reference was added for Branch Technical Position RLSB-1.
4-9	4.3	"multiplicable" was changed to "multiple".
4-9	4.3	In the last paragraph a comma was added to "Westinghouse plants, respectively, are listed…"
4-10	4.3	In the last paragraph "nuclear steam supply system (NSSS)" was changed to "NSSS" as the acronym was previously defined.
4-10	4.3	Paragraphs on the *Inspection Strategy for Westinghouse/CE Core Barrel Weld Sampling* were deleted.
4-11 through 4-15	Table 4-1	Component ID numbers were added for all expansion components within the Expansion Link column.
4-11	Table 4-1	Note 10 added to the Examination Method/Frequency column for the CRGT spacer castings entry, based on NRC SER, Section 3.1.3.1 (RAI 1)
4-11	Table 4-1	Note 12 added to the Examination Coverage column for the CRGT spacer castings entry, based on NRC SER, Section 3.2.1 (RAI 13).
4-11	Table 4-1	Note 10 added to the Examination Method/Frequency for the top and bottom retaining rings entry, based on NRC SER, Section 3.1.3.1 (RAI 1).

Page[1]	Section[1]	Description of Change
4-12	Table 4-1	A space was added between "jack" and "screw" within the Examination Coverage column for both the original locking devices (key ring, pin) entry and the modified locking devices (bolt locking cup, jackscrew locking cup, bolted block) entry.
4-13	Table 4-1	Note 11 added to the Examination Method/Frequency column for the baffle-to-former bolt entry, based on NRC SER, Section 3.1.3.1 (RAI 4).
4-13	Table 4-1	Note 10 added to the Examination Method/Frequency column for the baffle plates entry, based on NRC SER, Section 3.1.3.1 (RAI 1).
4-14	Table 4-1	Wording updated in the Effect (Mechanism) column for the locking devices, including locking welds, of the baffle-to-former bolts and internal baffle-to-baffle bolts entry, based on NRC SER, Section 3.1.3.2 (RAI 6).
4-14	Table 4-1	Note 10 added to the Examination Method/Frequency column for the locking devices, including locking welds, of the baffle-to-former bolts and internal baffle-to-baffle bolts entry, based on NRC SER, Section 3.1.3.1 (RAI 1).
4-15	Table 4-1	"Spider welds" was updated to "Spider Welds" in the Effect (Mechanism)" and "Examination Coverage" columns for the IMI guide tube spiders and IMI guide tube spider-to-lower grid rib section welds entry.
4-16	Table 4-1	Notes 10, 11, and 12 added to the table notes, based on NRC SER, Section 3.1.3.1 (Note 10 – RAI 1, Note 11 – RAI 4) and Section 3.2.1 (Note 12 – RAI 13).
4-17 through 4-20	Table 4-2	Component ID numbers were added for all expansion components within the Expansion Link column.
4-17, 4-18, and 4-20	Table 4-2	Clarification of ¾" adjacent base metal added to the Examination Coverage column for the core shroud plate-former plate weld, shroud plates, upper flange weld, middle girth weld, and deep beams based on NRC RAI 20.
4-18	Table 4-2	The figure for the Core Shroud Assembly (Welded) Assembly within the Examination Coverage column was updated to Figure 4-26.
4-18	Table 4-2	Examination Coverage of 100% of accessible weld identified for one side of the upper flange weld and the 100% of the accessible weld outer diameter of the middle girth weld based on NRC RAI 29.
4-18	Table 4-2	Core Support Columns added as an Expansion Link for the Middle Girth Weld based on NRC RAI 9.
4-19	Table 4-2	Clarification added to the Examination Method/Frequency column for the CSB flexure weld based on NRC RAI 16.
4-19	Table 4-2	Fatigue deleted from the Examination Coverage column for the CSB flexure weld.
4-20	Table 4-2	Remaining deep beams were added within the Expansion Link column for the deep beams, and clarification and notes 5, 6, and 7 added to the Examination Coverage column for the deep beams based on NRC RAI 10.
N/A	Table 4-2	Lower support structure core support columns were deleted and moved to the CE expansion table, Table 4-5 based on NRC RAI 9.
4-21	Table 4-2	Table note 4 edited based on NRC RAI 29.
4-21	Table 4-2	Table notes 6 and 7 added based on NRC RAI 10.
4-22 through 4-24	Table 4-3	Component ID numbers were added for all expansion components within the Expansion Link column.
4-22	Table 4-3	Clarification of ¾" adjacent base metal added to the Examination Coverage column for the upper flange weld and lower girth weld, based on NRC RAI 20.

Page[1]	Section[1]	Description of Change
4-22	Table 4-3	Examination Coverage of 100% of accessible weld identified for one side of the upper flange weld and the 100% of the accessible weld identified for the outer diameter of the lower girth weld based on NRC RAI 29.
4-22	Table 4-3	"(See NSAL-17-1)" was added to the "Applicability" column for the guide plates (cards).
4-23	Table 4-3	Clarifications added to the Examination Method/Frequency column for the baffle-former bolts and notes 8 and 9 added based on NRC RAI 8 and clarification on the response to RAI 8.
4-24	Table 4-3	Note 5 updated based on NRC RAI 19 and clarification on the response to RAI 19. Note 5 was also updated to include a reference to EPRI letter MRP 2018-007.
4-24	Table 4-3	Note 6 updated based on NRC RAI 29.
4-24	Table 4-3	Notes 8 and 9 added based on NRC RAI 8 and clarification on the response to RAI 8. Note 10 added to the Examination Coverage column for the thermal shield flexures component entry in order to reference the technical bulletin.
4-25	Table 4-4	Note 9 added to the Expansion Item column for the vent valve bodies entry, based on NRC SER, Section 3.1.3.1 (RAI 3).
4-25 and 4-27	Table 4-4	"B7, B8, B12. UCB, LCB, or FD bolts and their locking devices" was updated to "B7.UCB, B8.LCB, or B12.FD bolts and their locking devices" within the Primary Link (Note 1) column.
4-27	Table 4-4	Note 1 updated in the table notes based on NRC SER, Section 3.1.3.2 (RAI 17).
4-28	Table 4-4	Note 9 added to the table notes based on NRC SER, Section 3.1.3.1 (RAI 3)
4-29 and 4-30	Table 4-5	Clarification of ¾" adjacent base metal added to the Examination Coverage column for the lower girth weld, upper girth weld, upper axial weld, lower core support beams, middle axial weld and lower axial weld, remaining axial welds, remaining axial welds and ribs and rings based on NRC RAI 20.
4-29	Table 4-5	Note 5 added to the Examination Coverage column for the lower girth weld, upper girth weld, upper axial weld, middle axial weld, and lower axial weld and to the table notes based on NRC RAI 29.
4-29	Table 4-5	Examination Coverage of 100% of the accessible weld identified for the outer diameter of the lower girth weld, middle axial weld, and lower axial weld and 100% of the accessible weld identified for one side of the upper girth weld and upper axial weld based on NRC RAI 29.
4-30	Table 4-5	Clarification added to the Examination Method/Frequency column and the Examination Coverage column for the remaining axial welds and ribs and rings based on NRC RAI 12.
4-31	Table 4-5	Lower support structure core support columns were added as an expansion component in Table 4-5 and note 3 was added based on NRC RAI 9. Further examination coverage clarification and note 4 were added based on clarification on the response to NRC RAI 9.
4-32 and 4-33	Table 4-6	Clarification of ¾" adjacent base metal added to the Examination Coverage column for the upper girth weld, upper axial weld, lower flange weld, middle axial welds, and lower axial welds and 0.25 inch of the adjacent base metal added for the remaining CRGT lower flange welds based on NRC RAI 20.
4-32 and 4-33	Table 4-6	Examination Coverage of 100% of the accessible weld identified for one side of the upper girth weld and upper axial weld and 100% of the accessible weld identified for the outer diameter of the lower flange weld, middle axial welds, and lower axial welds based on NRC RAI 29. Notes 5 and 6 also added based on NRC RAI 29.

Page[1]	Section[1]	Description of Change
4-33	Table 4-6	Clarification added to the Examination Coverage column for the lower support forging or castings based on NRC RAI 14.
4-33	Table 4-6	Addition of note 3 to the Examination Coverage column for the lower support forging or castings and upper core plate based on NRC RAI 14.
4-33	Table 4-6	Clarification added to the Examination Coverage column for the lower support column bodies (both cast and non-cast) based on NRC RAI 9. Further examination coverage clarification and notes 3 and 4 were added based on clarification on the response to NRC RAI 9.
4-33	Table 4-6	Figure 4-22 added for the lower support column bolts in the Examination Coverage column.
4-34	Table 4-6	Note 3 added to the table notes based on NRC RAI 14.
4-34	Table 4-6	Note 4 added to the table notes based on clarification on the response to NRC RAI 9.
4-34	Table 4-6	Notes 5 and 6 added to the table notes based on NRC RAI 29.
4-35	4.4	Core Support Assembly added to paragraph 3 as an example of the core support structure. A comma was added to paragraph 3 in the sentence, "Thus, ASME Code…" In paragraph 4 "component specific" was updated to "component-specific".
4-36	4.4	In the first paragraph, "age related" was changed to "age-related".
4-36	4.4	In paragraph 2, "in-service" was changed to "inservice" in two places.
4-36	4.4	Per the response to RAI 15, the control rod guide tube support pin paragraph was removed and added to Section 4.5 and paragraph 5 was revised to state: "A plant-specific Westinghouse-design item that is managed within the plant program is guidance for flux thimble tubes. The bottom mounted instrumentation flux thimble tubes (applicable to all plants) are inspected to ensure pressure boundary integrity, as initially prompted by Bulletin IEB 88-09." Reference 45 was also added for IEB 88-09.
4-37	4.5	In the first paragraph, "in-service" was changed to "inservice" in two places.
4-37 and 4-38	4.5	Per the response to RAI 15, the control rod guide tube support pin text from Section 4.4 was added after the first paragraph of this section and updated.
4-69	N/A	The figure title for Figure 4-37 was changed from "CE-Design Core Shroud Assembly (Welded) Remaining Axial Welds, Ribs, and Rings" to "Typical CE-Design Core Shroud Assembly (Welded Full-Height) Core Shroud Assembly".
5-1	5	WCAP-17096-NP was updated to WCAP-17096-NP-A within second paragraph.
5-2 through 5-10	Table 5-1	Component ID numbers were added for all primary components within the Primary Item column and all expansion components within the Expansion Link(s) column.
5-7	Table 5-1	Primary Item Examination Acceptance Criteria column updated for the baffle plate entry based on NRC SER, Section 3.1.3.3 (RAI 22).
5-11 through 5-14	Table 5-2	Component ID numbers were added for all primary components within the Primary Item column and all expansion components within the Expansion Link(s) column.
5-11	Table 5-2	Clarification added to the Expansion Criteria column for the core shroud assembly (welded) shroud plates based on NRC RAI 12.
5-12	Table 5-2	VT-3 was changed to VT-1 within the Examination Acceptance Criteria column for the core shroud assembly (welded) assembly based on NRC RAI 23.
5-12 and 5-13	Table 5-2	The Expansion Criteria column was updated to remove the extended coverage of 100% for the upper flange weld and middle girth weld as 100% coverage of accessible weld is now required based on NRC RAI 29.

Page[1]	Section[1]	Description of Change
5-13	Table 5-2	Clarification on core support column requirements added to the Expansion Link(s) column, Expansion Criteria column, and Expansion Item Examination Acceptance Criteria column for the middle girth weld based on NRC RAI 9 and on clarification on the response to NRC RAI 9.
5-14	Table 5-2	Clarification added to the Expansion Link(s) column, Expansion Criteria column, and Expansion Item Examination Acceptance Criteria column for the deep beams based on NRC RAI 10.
5-15 through 5-18	Table 5-3	Component ID numbers were added for all primary components within the Primary Item column and all expansion components within the Expansion Link(s) column.
5-15	Table 5-3	Reference link added to WCAP-17451-P within the Expansion Item Examination Acceptance Criteria column for the guide plates (cards).
5-15	Table 5-3	Clarification added to the Expansion Criteria Column for the upper flange weld based on NRC RAI 14.
5-16	Table 5-3	Clarification added to the Expansion Criteria Column for the lower girth weld based on NRC RAI 14 and on clarification on the response to NRC RAI 9.
5-15 and 5-16	Table 5-3	The Expansion Criteria column was updated to remove the extended coverage of 100% for the upper flange weld and lower girth weld as 100% coverage of accessible weld is now required based on NRC RAI 29. Response to NRC RAI 29 supersedes the response to NRC RAI 14.
5-19	5.1.1	"completely separated material" was changed to "separated material" in paragraph 3.
5-20	5.1.3	"reactor internals" has been changed to "PWR internals" in first paragraph.
5-21	5.2	Section 4.3 in the first paragraph was updated to Section 4.2.5.
6-1	6	WCAP-17096-NP was updated to WCAP-17096-NP-A within third paragraph.
6-2	6	WCAP-17096-NP was updated to WCAP-17096-NP-A within first paragraph.
7-1	7	"reactor internals" has been changed to "PWR internals" in second paragraph.
7-2	7.3	The last two paragraphs of the Reactor Internals Guidelines Implementation Requirement section were updated to: "For units that have submitted an AMP to the regulator under MRP-227-A, and their period of extended operation begins no later than January 1, 2022, that MRP-227-A based program may be implemented as the baseline inspection without deviation from this Needed requirement. Requirements contained within the interim guidance letters applicable to that unit must also be incorporated in the plant program (e.g., MRP-2014-006, MRP-2013-023, MRP 2016-021, and MRP 2017-009). However, subsequent implementation shall be in accordance with revision 1-A of this guideline. Updates of engineering programs for this revision of the guideline for each reactor shall be implemented by January 1, 2022; however, these NRC-endorsed guidelines may be implemented immediately, once published."
7-2	7.5	WCAP-17096-NP was updated to WCAP-17096-NP-A within last sentence.
8-3	8	Reference 26 for WCAP-17096-NP was updated to WCAP-17096-NP-A.
8-3 and 8-4	8	References 35 through 46 added.
A-1	Appendix A	"Aging Management Program (AMP)" was updated to "AMP" as acronym was previously defined.

Page[1]	Section[1]	Description of Change
A-1	Appendix A	"have" has been changed to "has" in the third paragraph.
A-1	Appendix A	In the IGSCC paragraph, "in-service" was changed to "inservice".
A-2	Appendix A	Degree symbol was corrected for 1800°F in footnote 9.
A-4	Appendix A	In the last paragraph, "uninspectable" was changed to "not inspectable".
A-5 and A-6	Appendix A	Additional operating experience was added for Westinghouse-design plants in downflow configuration with Type 347SS baffle-former bolts.
A-6	Appendix A	Additional operating experience was added for the core support barrel welds for CE-designed plants and for the core barrel welds for Westinghouse-designed plants.
A-7	Appendix A	Additional operating experience sentence was added to discuss thermal shield operating experience in relation to IASCC degradation of the core support barrel.
A-8	Appendix A	Additional operating experience was added for Westinghouse-designed control rod guide tubes and control rod guide cards.
A-11	Appendix A	"AREVA (now Framatome)" was added to the first paragraph of the *Mechanism Unidentified to Date* section because of the company's name change since the publication of MRP-227 Revision 1.
B-2	Appendix B	Acronym definitions for Applicant/Licensee Action Items, Combustion Engineering, Electric Power Research Institute, Materials Reliability Program, Nuclear Regulatory Commission, and Inspection and Evaluation have been removed as these acronyms were previously defined.
B-2	Appendix B	Reference link for ML14309A484 was added within last sentence.
B-3	Appendix B	"reactor internals" has been changed to "PWR internals" in two places.
B-4	Appendix B	"re-inspection" was changed to "reinspection" in two places.
B-5	Appendix B	"re-inspection" was changed to "reinspection" within first paragraph. "cumulartive" was corrected to "cumulative" within the last paragraph under Upper Axial Boundary Limitations.
B-6	Appendix B	"reactor internals" has been changed to "PWR internals" in four places.
D-1	Appendix D	Appendix D was added to document the NRC RAIs on MRP-227 Revision 1.

Part 2 Table Notes:

1. Page numbers and section numbers correspond to this Revision, MRP-227 Revision 1-A.

Part 3 - MRP-227 Revisions 1 and 1-A Primary and Expansion Table Changes – Comparison from MRP-227-A

TABLE 4-1: B&W plants Primary Items

Item	Applicability	Effect (Mechanism)	Expansion Link	Exam. Meth./ Freq.	Exam. Coverage
Existing MRP-227-A entries **Plenum Cover Assembly & Core Support Shield Assembly** Plenum cover weldment rib pads Plenum cover support flange CSS top flange	All plants	Loss of material and associated loss of core clamping pre-load (Wear)	None	One-time physical measurement no later than two refueling outages from the beginning of the license renewal period. Perform subsequent visual (VT-3) examination on the 10-year ISI interval.	Determination of differential height of top of plenum rib pads to reactor vessel seating surface, with plenum in reactor vessel. See Figure 4-1.
MRP-227 Revision 1 entries **B1.Plenum Cover Assembly & Core Support Shield Assembly** a.Plenum cover weldment rib pads b.Plenum cover support flange c.Plenum cover support ring d.CSS top flange	NO CHANGE	NO CHANGE	NO CHANGE	One-time physical measurement no later than two refueling outages following entry into the period of extended operation. Subsequent visual (VT-3) examination prior to the end of each 10-year ISI interval.	Determination of differential height of top of plenum rib pads/plenum cover support ring location to reactor vessel seating surface, with plenum in reactor vessel. See Figure 4-1.
MRP-227 Revision 1-A entries **B1.Plenum Cover Assembly & Core Support Shield Assembly** a.Plenum cover weldment rib pads b.Plenum cover support flange c.Plenum cover support ring d.CSS top flange	Entry remained unchanged within MRP-227 Revision 1-A.				

TABLE 4-1: B&W plants Primary Items

Item	Applicability	Effect (Mechanism)	Expansion Link	Exam. Meth./ Freq.	Exam. Coverage
Existing MRP-227-A entries **Control Rod Guide Tube Assembly** CRGT spacer castings	All plants	Cracking (TE), including the detection of fractured spacers or missing screws	None	Visual (VT-3) examination during the next 10-year ISI. Subsequent examinations on the 10-year ISI interval.	Accessible surfaces at each of the 4 screw locations (at every 90°) of 100% of the CRGT spacer castings (limited accessibility). See Figure 4-5.
MRP-227 Revision 1 entries **B2.Control Rod Guide Tube Assembly** CRGT spacer castings	NO CHANGE	Cracking (TE), including the detection of fractured spacers or missing screws (Note 3)	Vent valve bodies	Visual (VT-3) examination during the next 10-year ISI interval. Subsequent examination prior to the end of each 10-year ISI interval.	Accessible surfaces at each of the four screw locations (at every 90°) of 100% of the CRGT spacer castings. (limited accessibility) See Figure 4-5.
MRP-227 Revision 1-A entries **B2.Control Rod Guide Tube Assembly** CRGT spacer castings	NO CHANGE	NO CHANGE	NO CHANGE	Visual (VT-3) examination during the next 10-year ISI interval. Subsequent examination prior to the end of each 10-year ISI interval. (Note 10)	Accessible surfaces at each of the four screw locations (at every 90°) of 100% of the CRGT spacer castings. (limited accessibility) (Note 12) See Figure 4-5.

TABLE 4-1: B&W plants Primary Items

Item	Applicability	Effect (Mechanism)	Expansion Link	Exam. Meth./ Freq.	Exam. Coverage
Existing MRP-227-A entries **Core Support Shield Assembly** CSS vent valve top retaining ring CSS vent valve bottom retaining ring (Note 1)	All plants	Cracking (TE), including the detection of surface irregularities, such as damaged, fractured material, or missing items	None	Visual (VT-3) examination during the next 10-year ISI. Subsequent examinations on the 10-year ISI interval.	100% of accessible surfaces (see BAW-2248A, page 4.3 and Table 4-1). See Figure 4-11.
MRP-227 Revision 1 entries **B3.Vent Valve Assembly** a.Vent valve top retaining ring b.Vent valve bottom retaining ring	NO CHANGE	Cracking (TE), including the detection of surface irregularities, such as damaged or fractured material in the retaining ring flanged region, or missing material (Note 3)	NO CHANGE	Visual (VT-3) examination during the next 10-year ISI interval. Subsequent examination during each 10-year ISI interval.	100% of accessible surfaces. (limited accessibility) See Figure 4-10.
MRP-227 Revision 1-A entries **B3.Vent Valve Assembly** a.Vent valve top retaining ring b.Vent valve bottom retaining ring	NO CHANGE	NO CHANGE	NO CHANGE	Visual (VT-3) examination during the next 10-year ISI interval. Subsequent examination during each 10-year ISI interval. (Note 10)	NO CHANGE

TABLE 4-1: B&W plants Primary Items

Item	Applicability	Effect (Mechanism)	Expansion Link	Exam. Meth./ Freq.	Exam. Coverage
Existing MRP-227-A entries **Core Support Shield Assembly** Upper core barrel (UCB) bolts and their locking devices	All plants	Bolts: Cracking (SCC) Locking Devices: Loss of material, damaged, distorted or missing locking devices (Wear or Fatigue damage by failed bolts).	UTS bolts and LTS studs/nuts or bolts and their locking devices SSHT studs/nuts or bolts and their locking devices (CR-3 and DB only) Lower grid shock pad bolts and their locking devices (TMI-1 only)	Volumetric examination (UT) of the bolts within two refueling outages from 1/1/2006 or next 10-year ISI interval, whichever is first. Subsequent examination on the 10-year ISI interval unless an evaluation of the baseline results submitted for NRC staff approval justifies a longer interval between examinations. Visual (VT-3) examination of bolt locking devices on the 10-year ISI interval.	100% of accessible bolts and their locking devices. (Note 3) See Figure 4-7.

TABLE 4-1: B&W plants Primary Items

Item	Applicability	Effect (Mechanism)	Expansion Link	Exam. Meth./ Freq.	Exam. Coverage
MRP-227 Revision 1 entries **B7.Core Support Shield Assembly** Upper core barrel (UCB) bolts and their locking devices	NO CHANGE	Bolts: Cracking (SCC– all bolts, Fatigue– original bolts only) Locking Devices: Loss of material, damaged or distorted or missing locking devices (Wear or Fatigue damage by failed bolts)	UTS bolts and LTS studs/nuts or bolts and their locking devices SSHT bolts and their locking devices (DB only)	Bolts: Volumetric (UT) examination during the next 10-year ISI interval. Locking Devices: Visual (VT-3) examination during the next 10-year ISI interval. Subsequent examination during each 10-year ISI interval unless an evaluation of the baseline results, submitted for NRC staff approval, justifies a longer interval between examinations.	100% of accessible bolts and their locking devices. (Note 6) See Figure 4-7.
MRP-227 Revision 1-A entries **B7.Core Support Shield Assembly** Upper core barrel (UCB) bolts and their locking devices	Entry remained unchanged within MRP-227 Revision 1-A.				

TABLE 4-1: B&W plants Primary Items

Item	Applicability	Effect (Mechanism)	Expansion Link	Exam. Meth./ Freq.	Exam. Coverage
Existing MRP-227-A entries					
Core Barrel Assembly Lower core barrel (LCB) bolts and their locking devices	All plants	Bolt: Cracking (SCC) Locking Devices: Loss of material, damaged, distorted or missing locking devices (Wear or Fatigue damage by failed bolts).	UTS bolts and LTS studs/nuts or bolts and their locking devices SSHT studs/nuts or bolts and their locking devices (CR-3 and DB only) Lower grid shock pad bolts and their locking devices (TMI-1 only)	Volumetric examination (UT) of the bolts during the next 10-year ISI interval from 1/1/2006. Subsequent examination on the 10-year ISI interval unless an evaluation of the baseline results submitted for NRC staff approval justifies a longer interval between examinations. Visual (VT-3) examination of bolt locking devices on the 10-year ISI interval.	100% of accessible bolts and their locking devices (Note 3) See Figure 4-8.

TABLE 4-1: B&W plants Primary Items

Item	Applicability	Effect (Mechanism)	Expansion Link	Exam. Meth./ Freq.	Exam. Coverage
MRP-227 Revision 1 entries **B8.Core Barrel Assembly** Lower core barrel (LCB) bolts and their locking devices	NO CHANGE	Bolts: Cracking (SCC, IC/ISR/Fatigue/Wear) (Note 7) Locking Devices: Loss of material, damaged or distorted or missing locking devices (Wear or Fatigue damage by failed bolts)	UTS bolts and LTS studs/nuts or bolts and their locking devices SSHT bolts and their locking devices (DB only)	Bolts: Volumetric (UT) examination during the next 10-year ISI interval. Locking Devices: Visual (VT-3) examination during the next 10-year ISI interval. Subsequent examination during each 10-year ISI interval unless an evaluation of the baseline results, submitted for NRC staff approval, justifies a longer interval between examinations.	100% of accessible bolts and their locking devices. (Note 6) See Figure 4-8(a).
MRP-227 Revision 1-A entries **B8.Core Barrel Assembly** Lower core barrel (LCB) bolts and their locking devices	Entry remained unchanged within MRP-227 Revision 1-A.				
Existing MRP-227-A entries **Core Barrel Assembly** Baffle-to-former bolts	All plants	Cracking (IASCC, IE, Overload) (Note 4)	Baffle-to-baffle bolts, Core barrel-to-former bolts	Baseline volumetric examination (UT) no later than two refueling outages from the beginning of the license renewal period with subsequent examination after 10 additional years.	100% of accessible bolts. (Note 3) See Figure 4-2.

TABLE 4-1: B&W plants Primary Items

Item	Applicability	Effect (Mechanism)	Expansion Link	Exam. Meth./ Freq.	Exam. Coverage
MRP-227 Revision 1 entries **B9.Core Barrel Assembly** Baffle-to-former bolts	All plants	Cracking (IASCC, IE, IC/ISR/Fatigue/Wear, Overload) (Notes 3, 7)	Baffle-to-baffle bolts Core barrel-to-former bolts	Volumetric (UT) examination during the next 10-year ISI interval. Subsequent examination during each 10-year ISI interval.	100% of accessible bolts. (Note 6) See Figure 4-2.
MRP-227 Revision 1-A entries **B9.Core Barrel Assembly** Baffle-to-former bolts	NO CHANGE	NO CHANGE	NO CHANGE	Volumetric (UT) examination during the next 10-year ISI interval. Subsequent examination during each 10-year ISI interval. (Note 11)	NO CHANGE
Existing MRP-227-A entries **Core Barrel Assembly** Baffle plates	All plants	Cracking (IE), including the detection of readily detectable cracking in the baffle plates	Core barrel cylinder (including vertical and circumferential seam welds), Former plates	Visual (VT-3) examination during the next 10-year ISI. Subsequent examinations on the 10-year ISI interval.	100% of the accessible surface within 1 inch around each flow and bolt hole. See Figure 4-2.

TABLE 4-1: B&W plants Primary Items

Item	Applicability	Effect (Mechanism)	Expansion Link	Exam. Meth./ Freq.	Exam. Coverage
MRP-227 Revision 1 entries **B10.Core Barrel Assembly** Baffle plates	All plants	Cracking (IE), including the detection of readily detectable cracking (Note 3)	Core barrel cylinder (including vertical and circumferential seam welds) Former plates Lower grid rib section	Visual (VT-3) examination during the next 10-year ISI interval. Subsequent examination during each 10-year ISI interval.	100% of the accessible surfaces within one inch around each flow and bolt hole. See Figure 4-2.
MRP-227 Revision 1-A entries **B10.Core Barrel Assembly** Baffle plates	NO CHANGE	NO CHANGE	NO CHANGE	Visual (VT-3) examination during the next 10-year ISI interval. Subsequent examination during each 10-year ISI interval. (Note 10)	NO CHANGE
Existing MRP-227-A entries **Core Barrel Assembly** Locking devices, including locking welds, of baffle-to-former bolts and internal baffle-to-baffle bolts	All plants	Cracking (IASCC, IE, Overload), including the detection of missing, nonfunctional, or removed locking devices or welds	Locking devices, including locking welds, for the external baffle-to-baffle bolts and Core barrel-to-former bolts	Visual (VT-3) examination during the next 10-year ISI. Subsequent examinations on the 10-year ISI interval.	100% of accessible baffle-to-former and internal baffle-to-baffle bolt locking devices. (Note 3) See Figure 4-2.

TABLE 4-1: B&W plants Primary Items

Item	Applicability	Effect (Mechanism)	Expansion Link	Exam. Meth./ Freq.	Exam. Coverage
MRP-227 Revision 1 entries **B11.Core Barrel Assembly** Locking devices, including locking welds, of baffle-to-former bolts and internal baffle-to-baffle bolts	All plants	Cracking (IASCC, IE), including the detection of missing, non-functional, or removed locking devices or welds (Note 3, 8)	Locking devices, including locking welds, of the external baffle-to-baffle bolts and core barrel-to-former bolts (Note 8)	Visual (VT-3) examination during the next 10-year ISI interval. Subsequent examination during each 10-year ISI interval.	100% of accessible baffle-to-former and internal baffle-to-baffle bolt locking devices and locking welds. (Note 6) See Figure 4-2.
MRP-227 Revision 1-A entries **B11.Core Barrel Assembly** Locking devices, including locking welds, of baffle-to-former bolts and internal baffle-to-baffle bolts	NO CHANGE	Locking Devices: Cracking (IASCC, IE), including the detection of missing, non-functional, or removed locking devices (Note 3) Locking Welds: Cracking (IE) including the detection of missing, non-functional, or removed locking device welds (Notes 3, 8)	NO CHANGE	Visual (VT-3) examination during the next 10-year ISI interval. Subsequent examination during each 10-year ISI interval. (Note 10)	NO CHANGE

TABLE 4-1: B&W plants Primary Items

Item	Applicability	Effect (Mechanism)	Expansion Link	Exam. Meth./ Freq.	Exam. Coverage
Existing MRP-227-A entries **Flow Distributor Assembly** Flow distributor (FD) bolts and their locking devices	All plants	Bolt: Cracking (SCC) Locking Devices: Loss of material, damaged or distorted or missing locking devices (Wear or Fatigue damage by failed bolts).	UTS bolts and LTS studs/nuts or bolts and their locking devices. SSHT studs/nuts or bolts and their locking devices (CR-3 and DB only) Lower grid shock pad bolts and their locking devices (TMI-1 only)	Volumetric examination (UT) of the bolts during the next 10-year ISI interval from 1/1/2006. Subsequent examination on the 10-year ISI interval unless an evaluation of the baseline results, submitted for NRC staff approval, justifies a longer interval between examinations. Visual (VT-3) examination of bolt locking devices on the 10-year ISI interval.	100% of accessible bolts and their locking devices. (Note 3) See Figure 4-8.

TABLE 4-1: B&W plants Primary Items

Item	Applicability	Effect (Mechanism)	Expansion Link	Exam. Meth./ Freq.	Exam. Coverage
MRP-227 Revision 1 entries **B12.Flow Distributor Assembly** Flow distributor (FD) bolts and their locking devices	All plants	Bolts: Cracking (SCC, Fatigue) Locking Devices: Loss of material, damaged or distorted or missing locking devices (Wear or Fatigue damage by failed bolts)	UTS bolts and LTS studs/nuts or bolts and their locking devices SSHT bolts and their locking devices (DB only)	Bolts: Volumetric (UT) examination during the next 10-year ISI interval. Locking Devices: Visual (VT-3) examination during the next 10-year ISI interval. Subsequent examination during each 10-year ISI interval unless an evaluation of the baseline results, submitted for NRC staff approval, justifies a longer interval between examinations.	100% of accessible bolts and their locking devices. (Note 6) See Figure 4-8(a).
MRP-227 Revision 1-A entries **B12.Flow Distributor Assembly** Flow distributor (FD) bolts and their locking devices	Entry remained unchanged within MRP-227 Revision 1-A.				
Existing MRP-227-A entries **Lower Grid Assembly** Alloy X-750 dowel-to-guide block welds	All plants	Cracking (SCC), including the detection of separated or missing locking welds, or missing dowels	Alloy X-750 dowel locking welds to the upper and lower grid fuel assembly support pads.	Initial visual (VT-3) examination no later than two refueling outages from the beginning of the license renewal period. Subsequent examinations on the 10-year ISI interval.	Accessible surfaces of 100% of the 24 dowel-to-guide block welds. See Figure 4-4.

TABLE 4-1: B&W plants Primary Items

Item	Applicability	Effect (Mechanism)	Expansion Link	Exam. Meth./ Freq.	Exam. Coverage
MRP-227 Revision 1 entries **B13.Lower Grid Assembly** Alloy X-750 dowel-to-guide block welds	All plants (except DB) (Note 9)	Cracking (SCC), including the detection of separated or missing locking welds, or missing dowels	Alloy X-750 dowel locking welds to the upper and lower grid fuel assembly support pads	Visual (VT-3) examination during the next 10-year ISI interval. Subsequent examination during each 10-year ISI interval.	Accessible surfaces of 100% of the dowel-to-guide block welds. See Figure 4-4.
MRP-227 Revision 1-A entries **B13.Lower Grid Assembly** Alloy X-750 dowel-to-guide block welds	Entry remained unchanged within MRP-227 Revision 1-A.				

TABLE 4-1: B&W plants Primary Items

Item	Applicability	Effect (Mechanism)	Expansion Link	Exam. Meth./ Freq.	Exam. Coverage
Existing MRP-227-A entries **Incore Monitoring Instrumentation** (IMI) Guide Tube Assembly IMI guide tube spiders IMI guide tube spider-to-lower grid rib section welds	All plants	Cracking (TE/IE), including the detection of fractured or missing spider arms or, Cracking (IE), including separation of spider arms from the lower grid rib section at the weld	Lower grid fuel assembly support pad items: pad, pad-to-rib section welds, Alloy X-750 dowel, cap screw, and their locking welds (Note: the pads, dowels, and cap screws are included because of IE of the welds)	Initial visual (VT-3) examination no later than two refueling outages from the beginning of the license renewal period. Subsequent examinations on the 10-year ISI interval.	100% of top surfaces of 52 spider castings and welds to the adjacent lower grid rib section. See Figures 4-3 and 4-6.
MRP-227 Revision 1 entries **B15.Incore Monitoring Instrumentation (IMI) Guide Tube Assembly** IMI guide tube spiders IMI guide tube spider-to-lower grid rib section welds	All plants	Spiders: Cracking (TE/IE), including the detection of fractured or missing spider arms (Note 3) Spider welds: Cracking (IE), including separation of spider arms from the lower grid rib section at the weld (Note 3)	Lower grid fuel assembly support pad component items: pad, pad-to-rib section welds, Alloy X-750 dowel, cap screws, and their locking welds (Note: the pads, dowels, and cap screws are included because of IE of the welds)	Visual (VT-3) examination during the next 10-year ISI interval. Subsequent examination during each 10-year ISI interval.	Spiders: 100% of the accessible top surfaces and 100% of the accessible spider surfaces adjacent to the spider casting welds. Spider welds: 100% of the accessible welds to the adjacent lower grid rib section. See Figures 4-3 and 4-6.

TABLE 4-1: B&W plants Primary Items

Item	Applicability	Effect (Mechanism)	Expansion Link	Exam. Meth./ Freq.	Exam. Coverage
MRP-227 Revision 1-A entries **B15.Incore Monitoring Instrumentation (IMI) Guide Tube Assembly** IMI guide tube spiders IMI guide tube spider-to-lower grid rib section welds	NO CHANGE	Spiders: Cracking (TE/IE), including the detection of fractured or missing spider arms (Note 3) Spider Welds: Cracking (IE), including separation of spider arms from the lower grid rib section at the weld (Note 3)	NO CHANGE	NO CHANGE	Spiders: 100% of the accessible top surfaces and 100% of the accessible spider surfaces adjacent to the spider casting welds. Spider Welds: 100% of the accessible welds to the adjacent lower grid rib section. See Figures 4-3 and 4-6.
Entry not Existing in MRP-227-A entries *The following items did not exist in the MRP-227-A table entries; items added as result of plant-specific review(s).*					
MRP-227 Revision 1 entries **B4.Vent Valve Assembly** Original locking devices (pressure plate, spring retainer, spring, U-cover)	Note 4	Loss of material from locking device (Wear associated with jack screw spring and pressure plate)	None	Visual (VT-3) examination during the next 10-year ISI interval (Note 5) Subsequent examination during each 10-year ISI interval.	100% of accessible surfaces, including the overall valve symmetry in the mounting ring and the overall jackscrew thread extension from the lower retaining ring threaded flange. See Figure 4-10.

TABLE 4-1: B&W plants Primary Items

Item	Applicability	Effect (Mechanism)	Expansion Link	Exam. Meth./ Freq.	Exam. Coverage
MRP-227 Revision 1-A entries **B4.Vent Valve Assembly** Original locking devices (pressure plate, spring retainer, spring, U-cover)	Entry remained unchanged within MRP-227 Revision 1-A.				
MRP-227 Revision 1 entries **B5.Vent Valve Assembly** Original locking devices (key ring, pin)	Note 4	Cracking (TE), including the detection of reportable indications	None	Visual (VT-3) examination during the next 10-year ISI interval. (Note 5) Subsequent examination during each 10-year ISI interval.	100% of accessible surfaces, including the overall valve symmetry in the mounting ring and the overall jackscrew thread extension from the lower retaining ring threaded flange. See Figure 4-10.
MRP-227 Revision 1-A entries **B5.Vent Valve Assembly** Original locking devices (key ring, pin)	Entry remained unchanged within MRP-227 Revision 1-A.				
MRP-227 Revision 1 entries **B6.Vent Valve Assembly** Modified locking devices (bolt locking cup, jackscrew locking cup, bolted block)	Note 4	Cracking (SCC), including detection of fractured or missing locking cups and welds, or the bolted block	None	Visual (VT-3) examination during the next 10-year ISI interval. (Note 5) Subsequent examination during each 10-year ISI interval.	100% of accessible surfaces, including the overall valve symmetry in the mounting ring and the overall jackscrew thread extension from the lower retaining ring threaded flange. See Figure 4-10.

TABLE 4-1: B&W plants Primary Items

Item	Applicability	Effect (Mechanism)	Expansion Link	Exam. Meth./ Freq.	Exam. Coverage
MRP-227 Revision 1-A entries **B6.Vent Valve Assembly** Modified locking devices (bolt locking cup, jackscrew locking cup, bolted block)	Entry remained unchanged within MRP-227 Revision 1-A.				
MRP-227 Revision 1 entries **B14.Lower Grid Assembly** Shock pad bolts and their locking devices	TMI-1	Bolts: Cracking (SCC, Fatigue) Locking Devices: Loss of material, damaged or distorted or missing locking devices (Wear or Fatigue damage by failed bolts)	UTS bolts and LTS bolts and their locking devices	Bolts: Volumetric (UT) examination during the next 10-year ISI interval. Locking Devices: Visual (VT-3) examination during the next 10-year ISI interval. Subsequent examination during each 10-year ISI interval unless an evaluation of the baseline results, submitted for NRC staff approval, justifies a longer interval between examinations.	100% of accessible bolts and their locking devices. (Note 6) See Figure 4-4.
MRP-227 Revision 1-A entries **B14.Lower Grid Assembly** Shock pad bolts and their locking devices	Entry remained unchanged within MRP-227 Revision 1-A.				

TABLE 4-1: B&W plants Primary Items

Existing MRP-227-A entries

Notes to Table 4-1:

1. A verification of the operation of each vent valve shall also be performed through manual actuation of the valve. Verify that the valves are not stuck in the open position and that no abnormal degradation has occurred. Examine the valves for evidence of scratches, pitting, embedded particles, leakage of the seating surfaces, cracking of lock welds and locking cups, jack screws for proper position, and wear. The frequency is defined in each unit's technical specifications or in their pump and valve inservice test programs (see BAW-2248A, page 4-3 and Table 4-1[18]).
2. Examination acceptance criteria and expansion criteria for the B&W components are in Table 5-1.
3. A minimum of 75% of the total population (examined + unexamined), including coverage consistent with the Expansion criteria in Table 5-1, must be examined for inspection credit.
4. The primary aging degradation mechanisms for loss of joint tightness for this item are IC and ISR. Fatigue and Wear, which can also lead to cracking, are secondary aging degradation mechanisms after significant stress relaxation and loss of preload has occurred due to IC/ISR. Bolt stress relaxation cannot readily be inspected by NDE. Only bolt cracking is inspected by UT inspection. The effect of loss of joint tightness on the functionality will be addressed by analysis of the core barrel assembly, which will be performed to address Applicant/Licensee action item 6 in the SE [27].

MRP-227 Revision 1 entries

Notes to Table 4-1:

1. Examination acceptance criteria and expansion criteria for the B&W component items and welds are detailed in Table 5-1.
2. Initial examinations may be scheduled concurrently with the next 10-year ISI but shall be performed no later than the final outage of the current 10-year ISI interval at the time of entry into the period of extended operation, except where otherwise noted.
3. Loss of ductility and fracture toughness, which are induced by thermal aging or neutron irradiation embrittlement, are not directly monitored. These effects are indirectly managed by performing periodic visual examinations capable of detecting cracking in the component.
4. As of May 2014, TMI-1 and DB have been verified to have original vent valve assembly locking devices only, while ONS-1, ONS-2, ONS-3, and ANO-1 have both original and modified vent valve assembly locking devices installed.
5. In addition to the Primary vent valve assembly visual (VT-3) examinations identified in this table, the following testing and inspection requirements are currently performed and will continue to be performed by the B&W units:

   A verification of the operation of each vent valve shall also be performed through manual actuation of the valve. Verify that the valves are not stuck in the open position and that no abnormal degradation has occurred. Examine the valves for evidence of leakage between the valve disc and the valve body (i.e., flow lines across the sealing surface), cracking of lock welds and locking cups, jackscrews for proper position, and wear. The frequency is defined in each unit's technical specifications or in their pump and valve inservice test programs (see BAW-2248A, page 4.3 and Table 4-1).
6. A minimum of 75% of the total population (examined + unexamined), including coverage consistent with the Expansion criteria in Section 5.3 of this document, must be examined for inspection credit.
7. The primary aging degradation mechanisms for loss of joint tightness for this item are IC and ISR. Fatigue and wear, which can also lead to cracking, are secondary aging degradation mechanisms after significant stress relaxation and loss of preload has occurred due to ISR/IC. Bolt stress relaxation cannot readily be inspected by NDE. Only bolt cracking is inspected by UT inspection.
8. The aging degradation mechanism of IASCC is only applicable to the baffle-to-former bolt and internal baffle-to-baffle bolt locking devices, not the baffle-to-former bolt and internal baffle-to-baffle bolt locking device welds. There are no Expansion component items for the baffle-to-former bolt and internal baffle-to-baffle bolt locking device welds for IASCC.
9. For the alternate configuration at DB, the difference in configuration shall be addressed on a unit-specific basis.

TABLE 4-1: B&W plants Primary Items

MRP-227 Revision 1-A entries

Notes to Table 4-1:

1. Examination acceptance criteria and expansion criteria for the B&W component items and welds are detailed in Table 5-1.
2. Initial examinations may be scheduled concurrently with the next 10-year ISI but shall be performed no later than the final outage of the current 10-year ISI interval at the time of entry into the period of extended operation, except where otherwise noted.
3. Loss of ductility and fracture toughness, which are induced by thermal aging or neutron irradiation embrittlement, are not directly monitored. These effects are indirectly managed by performing periodic visual examinations capable of detecting cracking in the component.
4. As of May 2014, TMI-1 and DB have been verified to have original vent valve assembly locking devices only, while ONS-1, ONS-2, ONS-3, and ANO-1 have both original and modified vent valve assembly locking devices installed.
5. In addition to the Primary vent valve assembly visual (VT-3) examinations identified in this table, the following testing and inspection requirements are currently performed and will continue to be performed by the B&W units:

   A verification of the operation of each vent valve shall also be performed through manual actuation of the valve. Verify that the valves are not stuck in the open position and that no abnormal degradation has occurred. Examine the valves for evidence of leakage between the valve disc and the valve body (i.e., flow lines across the sealing surface), cracking of lock welds and locking cups, jackscrews for proper position, and wear. The frequency is defined in each unit's technical specifications or in their pump and valve inservice test programs (see BAW-2248A, page 4.3 and Table 4-1).
6. A minimum of 75% of the total population (examined + unexamined), including coverage consistent with the Expansion criteria in Section 5.3 of this document, must be examined for inspection credit.
7. The primary aging degradation mechanisms for loss of joint tightness for this item are IC and ISR. Fatigue and wear, which can also lead to cracking, are secondary aging degradation mechanisms after significant stress relaxation and loss of preload has occurred due to ISR/IC. Bolt stress relaxation cannot readily be inspected by NDE. Only bolt cracking is inspected by UT inspection.
8. The aging degradation mechanism of IASCC is only applicable to the baffle-to-former bolt and internal baffle-to-baffle bolt locking devices, not the baffle-to-former bolt and internal baffle-to-baffle bolt locking device welds. There are no Expansion component items for the baffle-to-former bolt and internal baffle-to-baffle bolt locking device welds for IASCC.
9. For the alternate configuration at DB, the difference in configuration shall be addressed on a unit-specific basis.
10. The term "10-year ISI interval" is intended to mean the plant's existing schedule associated with removal of the core barrel, i.e., during the next scheduled 10-year ISI interval examination. As defined in the ASME B&PV Code Section XI, the first inspection interval is defined as 10 years following the initial start of plant commercial startup, and successive inspection intervals are defined as 10 years following the previous inspection interval. Each inspection interval may be extended by as much as 1 year and may be reduced without restriction, provided the examinations required for the interval have been completed. Successive intervals shall not extend more than 1 year beyond the original pattern of 10 year intervals and shall not exceed 11 years in length.
11. This assumes that all units operating as of December 2011 have performed baseline (initial) volumetric (UT) examinations no later than two refueling outages from the beginning of their first license renewal period.
12. At one of the CRGT locations at ANO-1, a device was installed to monitor water level that required removal of the center control rod drive mechanism and installation of an assembly that blocks access to all of the CRGT spacer castings except the top spacer casting. Therefore, for all plants except ANO-1, 100% of the CRGT spacer castings refers to 690 CRGT spacer castings. There is no requirement to examine this set of CRGT spacer castings at this one CRGT assembly at ANO-1 and 100% is defined as 680 CRGT spacer castings for ANO-1.

TABLE 4-2: CE plants Primary Items

Item	Applicability	Effect (Mechanism)	Expansion Link	Exam. Meth./ Freq.	Exam. Coverage
Existing MRP-227-A entries **Core Shroud Assembly** (Bolted) Core shroud bolts	Bolted plant designs	Cracking (IASCC, Fatigue) Aging Management (IE and ISR) (Note 2)	Core support column bolts, Barrel-shroud bolts	Baseline volumetric (UT) examination between 25 and 35 EFPY, with subsequent examination on a ten-year interval.	100% of accessible bolts (see Note 3). Heads are accessible from the core side. UT accessibility may be affected by complexity of head and locking device designs. See Figure 4-24.
MRP-227 Revision 1 entries **C1.Core Shroud Assembly (Bolted)** Core shroud bolts	Bolted plant designs	Cracking (IASCC, Fatigue) Aging Management (IE and ISR)	Core support column bolts, Barrel-shroud bolts	Baseline volumetric (UT) examination between 25 and 35 EFPY, with subsequent examination on a ten-year interval.	100% of accessible bolts (see Note 2). See Figure 4-25.
MRP-227 Revision 1-A entries **C1.Core Shroud Assembly (Bolted)** Core shroud bolts	Entry remained unchanged within MRP-227 Revision 1-A.				
Existing MRP-227-A entries **Core Shroud Assembly (Welded)** Core shroud plate-former plate weld	Plant designs with core shrouds assembled in two vertical sections	Cracking (IASCC) Aging Management (IE) (Note 2)	Remaining axial welds	Enhanced visual (EVT-1) examination no later than 2 refueling outages from the beginning of the license renewal period and subsequent examination on a ten-year interval.	Axial and horizontal weld seams at the core shroud reentrant corners as visible from the core side of the shroud, within six inches of central flange and horizontal stiffeners. See Figures 4-12 and 4-14.

TABLE 4-2: CE plants Primary Items

Item	Applicability	Effect (Mechanism)	Expansion Link	Exam. Meth./ Freq.	Exam. Coverage
MRP-227 Revision 1 entries **C2.Core Shroud Assembly (Welded)** Core shroud plate-former plate weld	Plant designs with core shrouds assembled in two vertical sections	Cracking (IASCC) Aging Management (IE)	Remaining axial welds	Enhanced visual (EVT-1) examination no later than 2 refueling outages from the beginning of the license renewal period and subsequent examination on a ten-year interval.	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 central flange and horizontal stiffeners. See Figure 4-26.
MRP-227 Revision 1-A entries **C2.Core Shroud Assembly (Welded)** Core shroud plate-former plate weld	Plant designs with core shrouds assembled in two vertical sections	Cracking (IASCC) Aging Management (IE)	C2.1.Remaining axial welds	Enhanced visual (EVT-1) examination no later than 2 refueling outages from the beginning of the license renewal period and subsequent examination on a ten-year interval.	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 central flange and horizontal stiffeners and ¾" of adjacent base metal. See Figure 4-26.
Existing MRP-227-A entries **Core Shroud Assembly (Welded)** Shroud plates	Plant designs with core shrouds assembled with full-height shroud plates	Cracking (IASCC) Aging Management (IE) (Note 2)	Remaining axial welds, Ribs and rings	Enhanced visual (EVT-1) examination no later than 2 refueling outages from the beginning of the license renewal period and subsequent examination on a ten-year interval.	Axial weld seams at the core shroud reentrant corners, at the core mid-plane (± three feet in height) as visible from the core side of the shroud. See Figure 4-13.

TABLE 4-2: CE plants Primary Items

Item	Applicability	Effect (Mechanism)	Expansion Link	Exam. Meth./ Freq.	Exam. Coverage
MRP-227 Revision 1 entries **C3.Core Shroud Assembly (Welded)** Shroud plates	Only plants with welded core shrouds assembled with full-height shroud plates	Cracking (IASCC) Aging Management (IE)	Remaining axial welds, Ribs and rings	Enhanced visual (EVT-1) examination no later than 2 refueling outages from the beginning of the license renewal period and subsequent examination on a ten-year interval.	Axial weld seams at the core shroud re-entrant corners, at the core mid-plane (± three feet in height) as visible from the core side of the shroud. See Figure 4-27.
MRP-227 Revision 1-A entries **C3.Core Shroud Assembly (Welded)** Shroud plates	Only plants with welded core shrouds assembled with full-height shroud plates	Cracking (IASCC) Aging Management (IE)	C3.1.Remaining axial welds, C3.2.Ribs and rings	Enhanced visual (EVT-1) examination no later than 2 refueling outages from the beginning of the license renewal period and subsequent examination on a ten-year interval.	Axial weld seams at the core shroud re-entrant corners, at the core mid-plane (± three feet in height) as visible from the core side of the shroud and ¾" of adjacent base metal. See Figure 4-27.
Existing MRP-227-A entries **Core Shroud Assembly** (Bolted) Assembly	Bolted plant designs	Distortion (Void Swelling), including: • Abnormal interaction with fuel assemblies • Gaps along high fluence shroud plate joints • Vertical displacement of shroud plates near high fluence joint Aging Management (IE)	None	Visual (VT-3) examination no later than 2 refueling outages from the beginning of the license renewal period. Subsequent examinations on a ten-year interval.	Core side surfaces as indicated. See Figures 4-25 and 4-26.

TABLE 4-2: CE plants Primary Items

Item	Applicability	Effect (Mechanism)	Expansion Link	Exam. Meth./ Freq.	Exam. Coverage
MRP-227 Revision 1 entries **C4.Core Shroud Assembly (Bolted)** Assembly	Bolted plant designs	Distortion (Void Swelling), including: • Abnormal interaction with fuel assemblies • Gaps along shroud plate joints • Relative Vertical displacement of shroud plates Aging Management (IE)	None	Visual (VT-3) examination no later than 2 refueling outages from the beginning of the license renewal period. Subsequent examinations on a ten-year interval.	Core side surfaces: • High fluence shroud plate joints • Top and bottom alignment of shroud plates • Bolts and locking devices See Figure 4-28.
MRP-227 Revision 1-A entries **C4.Core Shroud Assembly (Bolted)** Assembly	Entry remained unchanged within MRP-227 Revision 1-A.				
Existing MRP-227-A entries **Core Shroud Assembly (Welded)** Assembly	Plant designs with core shrouds assembled in two vertical sections	Distortion (Void Swelling), as evidenced by separation between the upper and lower core shroud segments Aging Management (IE)	None	Visual (VT-1) examination no later than 2 refueling outages from the beginning of the license renewal period. Subsequent examinations on a ten-year interval.	If a gap exists, make three to five measurements of gap opening from the core side at the core shroud re-entrant corners. Then, evaluate the swelling on a plant-specific basis to determine frequency and method for additional examinations. See Figures 4-12 and 4-14.

TABLE 4-2: CE plants Primary Items

Item	Applicability	Effect (Mechanism)	Expansion Link	Exam. Meth./ Freq.	Exam. Coverage
MRP-227 Revision 1 entries **C4a.Core Shroud Assembly (Welded)** Assembly	Plant designs with core shrouds assembled in two vertical sections	Distortion (Void Swelling), as evidenced by measurable separation between the upper and lower core shroud segments Aging Management (IE)	None	Visual (VT-1) examination no later than 2 refueling outages from the beginning of the license renewal period. Subsequent examinations on a ten-year interval.	100% of the horizontal seam between the upper and lower core shroud segments. (Note 4) See Figure 4-28.
MRP-227 Revision 1-A entries **C4a.Core Shroud Assembly (Welded)** Assembly	Plant designs with core shrouds assembled in two vertical sections	Distortion (Void Swelling), as evidenced by measurable separation between the upper and lower core shroud segments Aging Management (IE)	None	Visual (VT-1) examination no later than 2 refueling outages from the beginning of the license renewal period. Subsequent examinations on a ten-year interval.	100% of the horizontal seam between the upper and lower core shroud segments. (Note 3) See Figure 4-26.
Existing MRP-227-A entries **Core Support Barrel Assembly** Upper (core support barrel) flange weld	All plants	Cracking (SCC)	Lower core support Beams Core support barrel Assembly, upper cylinder Upper core barrel flange	Enhanced visual (EVT-1) examination no later than 2 refueling outages from the beginning of the license renewal period. Subsequent examinations on a ten-year interval.	100% of the accessible surfaces of the upper flange weld (Note 4). See Figure 4-15.

TABLE 4-2: CE plants Primary Items

Item	Applicability	Effect (Mechanism)	Expansion Link	Exam. Meth./ Freq.	Exam. Coverage
MRP-227 Revision 1 entries **C5.Core Support Barrel Assembly** Upper flange weld (UFW)	All plants	Cracking (SCC)	Upper Girth Weld (UGW), Lower Girth/Flange Weld (LGW/LFW), Upper Axial Welds (UAW), Lower core support beams	Enhanced visual (EVT-1) examination no later than 2 refueling outages from the beginning of the license renewal period. Subsequent examinations on a ten-year interval.	A minimum of 25% of the circumference of the UFW and adjacent base metal shall be examined. (Notes 5 and 6) See Figure 4-29.
MRP-227 Revision 1-A entries **C5.Core Support Barrel Assembly** Upper flange weld (UFW)	All plants	Cracking (SCC)	C5.2.Upper Girth Weld (UGW), C5.1.Lower Girth/Flange Weld (LGW/LFW), C5.3.Upper Axial Welds (UAW), C5.4.Lower core support beams	Enhanced visual (EVT-1) examination no later than 2 refueling outages from the beginning of the license renewal period. Subsequent examinations on a ten-year interval.	100% of the accessible weld length of one side of the UFW and ¾" of adjacent base metal shall be examined. (Note 4) See Figure 4-29.
Existing MRP-227-A entries **Core Support Barrel Assembly** Lower cylinder girth welds	All plants	Cracking (SCC, IASCC) Aging Management (IE)	Lower cylinder axial welds	Enhanced visual (EVT-1) examination no later than 2 refueling outages from the beginning of the license renewal period. Subsequent examinations on a ten-year interval.	100% of the accessible surfaces of the lower cylinder welds (Note 4). See Figure 4-15

TABLE 4-2: CE plants Primary Items

Item	Applicability	Effect (Mechanism)	Expansion Link	Exam. Meth./ Freq.	Exam. Coverage
MRP-227 Revision 1 entries **C6.Core Support Barrel Assembly** Middle Girth Weld (MGW)	All plants	Cracking (SCC, IASCC) Aging Management (IE)	Middle Axial Weld (MAW), Lower Axial Weld (LAW)	Enhanced visual (EVT-1) examination no later than 2 refueling outages from the beginning of the license renewal period. Subsequent examinations on a ten-year interval.	A minimum of 25% of the OD circumference of the MGW and adjacent base metal shall be examined. Notes (5 and 6) See Figure 4-29.
MRP-227 Revision 1-A entries **C6.Core Support Barrel Assembly** Middle Girth Weld (MGW)	All plants	Cracking (SCC, IASCC) Aging Management (IE)	C6.1.Middle Axial Weld (MAW), C6.2.Lower Axial Weld (LAW), C6.3.Core Support Columns	Enhanced visual (EVT-1) examination no later than 2 refueling outages from the beginning of the license renewal period. Subsequent examinations on a ten-year interval.	100% of the accessible weld length of the OD of the MGW and ¾" of adjacent base metal shall be examined. (Note 4) See Figure 4-29.
Existing MRP-227-A entries **Core Support Barrel Assembly** Lower flange weld	All plants	Cracking (Fatigue)	None	If fatigue life cannot be demonstrated by time-limited aging analysis (TLAA), 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.	Examination coverage to be defined by evaluation to determine the potential location and extent of fatigue cracking. See Figures 4-15 and 4-16.

TABLE 4-2: CE plants Primary Items

Item	Applicability	Effect (Mechanism)	Expansion Link	Exam. Meth./ Freq.	Exam. Coverage
MRP-227 Revision 1 entries **C7.Core Support Barrel Assembly** CSB Flexure Weld (CSBFW)	All plants with welded core shrouds	Cracking (Fatigue, SCC)	None	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.	Examination coverage to be defined by evaluation to determine the potential location and extent of fatigue cracking. See Figure 4-30.
MRP-227 Revision 1-A entries **C7.Core Support Barrel Assembly** CSB Flexure Weld (CSBFW)	All plants with welded core shrouds	Cracking (Fatigue, SCC)	None	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.	Examination coverage to be defined by evaluation to determine the potential location and extent of cracking. See Figure 4-30.

TABLE 4-2: CE plants Primary Items

Item	Applicability	Effect (Mechanism)	Expansion Link	Exam. Meth./ Freq.	Exam. Coverage
Existing MRP-227-A entries **Lower Support Structure** Core support column welds	All plants	Cracking (SCC, IASCC, Fatigue including damaged or fractured material) Aging Management (IE, TE)	None	Visual (VT-3) examination no later than 2 refueling outages from the beginning of the license renewal period. Subsequent examinations on a ten-year interval.	100% of the accessible surfaces of the core support column welds (Note 5). See Figures 4-16 and 4-31
MRP-227 Revision 1 entries **C8.Lower Support Structure** Core support columns	Plants with full-height bolted or half-height welded core shroud plates	Cracking (SCC, IASCC, Fatigue including damaged or fractured material) Aging Management (IE, TE)	None	Visual (VT-3) examination no later than 2 refueling outages from the beginning of the license renewal period. Subsequent examinations on a ten-year interval.	Plants with full height bolted core shroud plates: 25% of column assemblies as visible using a VT-3 examination from above the lower core plate. Plants with core shrouds assembled in two vertical sections: 100% of the accessible surfaces of the core support column welds, from the top side of the core support plate (Note 3). See Figure 4-31.
MRP-227 Revision 1-A entries **C8.Lower Support Structure** Core support columns	The lower support structure core support columns were moved to expansion table 4-5.				

TABLE 4-2: CE plants Primary Items

Item	Applicability	Effect (Mechanism)	Expansion Link	Exam. Meth./ Freq.	Exam. Coverage
Existing MRP-227-A entries **Lower Support Structure** Core support plate	All plants with a core support plate	Cracking (Fatigue) Aging Management (IE)	None	If fatigue life cannot be demonstrated by time-limited aging analysis (TLAA), 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.	Examination coverage to be defined by evaluation to determine the potential location and extent of fatigue cracking. See Figure 4-16.
MRP-227 Revision 1 entries **C9.Lower Support Structure** Core support plate	All plants with a core support plate	Cracking (Fatigue) Aging Management (IE)	None	If screening for fatigue cannot be satisfied by plant-specific evaluation, 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.	Examination coverage to be defined by evaluation to determine the potential location and extent of fatigue cracking. See Figure 4-31.
MRP-227 Revision 1-A entries **C9.Lower Support Structure** Core support plate	Entry remained unchanged within MRP-227 Revision 1-A.				

TABLE 4-2: CE plants Primary Items

Item	Applicability	Effect (Mechanism)	Expansion Link	Exam. Meth./ Freq.	Exam. Coverage
Existing MRP-227-A entries **Upper Internals Assembly** Fuel alignment plate	All plants with core shrouds assembled with full-height shroud plates	Cracking (Fatigue)	None	If fatigue life cannot be demonstrated by time-limited aging analysis (TLAA), 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.	Examination coverage to be defined by evaluation to determine the potential location and extent of fatigue cracking. See Figure 4-17.
MRP-227 Revision 1 entries **C10.Upper Internals Assembly** Fuel alignment plate	Only plants with welded core shrouds assembled with full-height shroud plates	Cracking (Fatigue), Aging management (IE)	None	If screening for fatigue cannot be satisfied by plant-specific evaluation, 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.	Examination coverage to be defined by evaluation to determine the potential location and extent of fatigue cracking. See Figure 4-32.
MRP-227 Revision 1-A entries **C10.Upper Internals Assembly** Fuel alignment plate	Entry remained unchanged within MRP-227 Revision 1-A.				

TABLE 4-2: CE plants Primary Items

Item	Applicability	Effect (Mechanism)	Expansion Link	Exam. Meth./ Freq.	Exam. Coverage
Existing MRP-227-A entries **Control Element Assembly** Instrument guide tubes	All plants with instrument guide tubes in the CEA shroud assembly	Cracking (SCC, Fatigue) that results in missing supports or separation at the welded joint between the tubes and supports	Remaining instrument guide tubes within the CEA shroud assemblies	Visual (VT-3) examination, no later than 2 refueling outages from the beginning of the license renewal period. Subsequent examination on a ten-year interval. Plant-specific component integrity assessments may be required if degradation is detected and remedial action is needed.	100% of tubes in peripheral CEA shroud assemblies (i.e., those adjacent to the perimeter of the fuel alignment plate). See Figure 4-18.
MRP-227 Revision 1 entries **C11.Control Element Assembly** Instrument guide tubes	All plants with instrument guide tubes attached to the CEA shroud assemblies	Cracking (SCC, Fatigue) that results in missing supports or separation at the welded joint between the tubes and supports	Remaining instrument guide tubes within the CEA shroud assemblies	Visual (VT-3) examination, no later than 2 refueling outages from the beginning of the license renewal period. Subsequent examination on a ten-year interval.	100% of instrument guide tubes in peripheral CEA shroud assemblies (i.e., those adjacent to the perimeter of the fuel alignment plate), focusing on the supports and the connecting welds between the supports and the guide tube and CEA shroud. See Figure 4-33.

TABLE 4-2: CE plants Primary Items

Item	Applicability	Effect (Mechanism)	Expansion Link	Exam. Meth./ Freq.	Exam. Coverage
MRP-227 Revision 1-A entries **C11.Control Element Assembly** Instrument guide tubes	Entry remained unchanged within MRP-227 Revision 1-A.				
Existing MRP-227-A entries **Lower Support Structure** Deep beams	All plants with core shrouds assembled with full-height shroud plates	Cracking (Fatigue) that results in a detectable surface-breaking indication in the welds or beams Aging Management (IE)	None	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 adequacy of remaining fatigue life cannot be demonstrated.	Examine beam-to-beam welds, in the axial elevation from the beam top surface to four inches below. See Figure 4-19.
MRP-227 Revision 1 entries **C12.Lower Support Structure** Deep beams	Only plants with welded core shrouds assembled with full-height shroud plates	Cracking (Fatigue) that results in a detectable surface-breaking indication in the welds or beams. Aging Management (IE)	None	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 adequacy of remaining fatigue life cannot be demonstrated.	Examine 25% of the total number of beam-to-beam welds. Coverage on each weld includes the top four inches of the weld in the vertical orientation. See Figure 4-34.

TABLE 4-2: CE plants Primary Items

Item	Applicability	Effect (Mechanism)	Expansion Link	Exam. Meth./ Freq.	Exam. Coverage
MRP-227 Revision 1-A entries **C12.Lower Support Structure** Deep beams	Only plants with welded core shrouds assembled with full-height shroud plates	Cracking (Fatigue) that results in a detectable surface-breaking indication in the welds or beams. Aging Management (IE)	Remaining deep beams	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 adequacy of remaining fatigue life cannot be demonstrated.	Examine 25% of the total number of beam-to-beam welds. Coverage on each weld includes the top four inches of the weld in the vertical orientation and ¾" of adjacent base metal. The inspection coverage must be evenly distributed across the deep beam structure. (Notes 5, 6, and 7) See Figure 4-33.

Existing MRP-227-A entries

Note to Table 4-2:

1. Examination acceptance criteria and expansion criteria for the CE components are in Table 5-2.
2. Void swelling effects on this component is managed through management of void swelling on the entire core shroud assembly.
3. A minimum of 75% of the total population (examined + unexamined), including coverage consistent with the Expansion criteria in Table 5-2, must be examined for inspection credit.
4. A minimum of 75% of the total weld length (examined + unexamined), including coverage consistent with the Expansion criteria in Table 5-2, must be examined from either the inner or outer diameter for inspection credit.
5. A minimum of 75% of the total population of core support column welds.

MRP-227 Revision 1 entries

Notes to Table 4-2:

1. Examination acceptance criteria and expansion criteria for the CE components are in Table 5-2.
2. A minimum of 75% of the total core shroud bolts population (examined + unexamined), including coverage consistent with the Expansion criteria in Table 5-2, must be examined for inspection credit.
3. A minimum of 25% of the total population of core support column welds
4. If evidence of distortion is detected by visual exam, consideration should be given to making supplementary measurements (minimum of 3 to 5, depending on extent of observed condition) of gap opening between the upper and lower core shroud segments.
5. Examination coverage requires a minimum of 25% of the circumference of either the ID or the OD of the weld.
6. The stated coverage requirement is the minimum if no significant indications are found. However the Examination Acceptance criteria in Section 5 require that additional coverage must be achieved in the same outage if significant flaws are found. This contingency should be considered for inspection planning purposes.

TABLE 4-2: CE plants Primary Items

Item	Applicability	Effect (Mechanism)	Expansion Link	Exam. Meth./ Freq.	Exam. Coverage

MRP-227 Revision 1-A entries

Notes to Table 4-2:

1. Examination acceptance criteria and expansion criteria for the CE components are in Table 5-2.
2. A minimum of 75% of the total core shroud bolts population (examined + unexamined), including coverage consistent with the Expansion criteria in Table 5-2, must be examined for inspection credit.
3. If evidence of distortion is detected by visual exam, consideration should be given to making supplementary measurements (minimum of 3 to 5, depending on extent of observed condition) of gap opening between the upper and lower core shroud segments.
4. Examination coverage requires a minimum of 75% of the length of either the ID or the OD of the weld being examined.
5. The stated coverage requirement is the minimum if no significant indications are found. However the Examination Acceptance criteria in Section 5 require that additional coverage must be achieved in the same outage if significant flaws are found. This contingency should be considered for inspection planning purposes.
6. Inspections subsequent to the first 25% sampling inspection must cover weld length that was not inspected during previous examinations. A minimum of 20% out of the 25% inspection coverage of each subsequent inspection must include previously un-inspected weld length as possible within the limitation of the remaining accessible un-inspected weld.
7. Justification that adequate distribution of the inspection coverage has been achieved can be based on geometric or layout arguments. For example, since each rectangular axial compartment in the structure has four welds around it, which are shared with the adjacent compartments on two sides, the inspection coverage could consist of the selection of one weld from every second compartment in the structure.

TABLE 4-3: Westinghouse plants Primary Items

Item	Applicability	Effect (Mechanism)	Expansion Link	Exam. Meth./ Freq.	Exam. Coverage
Existing MRP-227-A entries **Control Rod Guide Tube Assembly** Guide plates (cards)	All plants	Loss of Material (Wear)	None	Visual (VT-3) examination no later than 2 refueling outages from the beginning of the license renewal period, and no earlier than two refueling outages prior to the start of the license renewal period. Subsequent examinations are required on a ten-year interval.	20% examination of the number of CRGT assemblies, with all guide cards within each selected CRGT assembly examined. See Figure 4-20
MRP-227 Revision 1 entries **W1.Control Rod Guide Tube Assembly** Guide plates (cards)	All plants	Loss of Material (Wear)	None	Per the requirements of WCAP-17451-P, including subsequent examinations (Note 7).	Examination coverage per the requirements of WCAP-17451-P, Revision 1 (Note 7) See Figure 4-11
MRP-227 Revision 1-A entries **W1.Control Rod Guide Tube Assembly** Guide plates (cards)	All plants	Loss of Material (Wear)	None	Per the requirements of WCAP-17451-P, including subsequent examinations (Note 5).	Examination coverage per the requirements of WCAP-17451-P, Revision 1 (Note 5) See Figure 4-11

TABLE 4-3: Westinghouse plants Primary Items

Item	Applicability	Effect (Mechanism)	Expansion Link	Exam. Meth./ Freq.	Exam. Coverage
Existing MRP-227-A entries **Control Rod Guide Tube Assembly** Lower flange welds	All plants	Cracking (SCC, Fatigue) Aging Management (IE and TE)	Bottom-mounted instrumentation (BMI) column bodies Lower support column bodies (cast) Upper core plate Lower support forging/casting	Enhanced visual (EVT-1) examination to determine the presence of crack-like surface flaws in flange welds no later than 2 refueling outages from the beginning of the license renewal period and subsequent examination on a ten-year interval.	100% of outer (accessible) CRGT lower flange weld surfaces and adjacent base metal on the individual periphery CRGT assemblies. (Note 2) See Figure 4-21.
MRP-227 Revision 1 entries **W2.Control Rod Guide Tube Assembly** Lower flange welds	All plants	Cracking (SCC, Fatigue) Aging Management (IE and TE)	Remaining CRGT assembly lower flange welds BMI column bodies	Enhanced visual (EVT-1) examination to determine the presence of crack-like surface flaws in flange welds no later than 2 refueling outages from the beginning of the license renewal period and subsequent examination on a ten-year interval.	100% of outer (accessible) CRGT lower flange weld surfaces and 0.25-inch of the adjacent base metal on the individual peripheryCRGT assemblies.(Note 2) See Figure 4-12.
MRP-227 Revision 1-A entries **W2.Control Rod Guide Tube Assembly** Lower flange welds	Entry remained unchanged within MRP-227 Revision 1-A.				

TABLE 4-3: Westinghouse plants Primary Items

Item	Applicability	Effect (Mechanism)	Expansion Link	Exam. Meth./ Freq.	Exam. Coverage
Existing MRP-227-A entries **Core Barrel Assembly** Upper core barrel flange weld	All plants	Cracking (SCC)	Lower support column bodies (non-cast) Core barrel outlet nozzle welds	Periodic enhanced visual (EVT-1) examination, no later than 2 refueling outages from the beginning of the license renewal period and subsequent examination on a ten-year interval.	100% of one side of the accessible surfaces of the selected weld and adjacent base metal (Note 4). See Figure 4-22.
MRP-227 Revision 1 entries **W3.Core Barrel Assembly** Upper flange Weld (UFW)	All plants	Cracking (SCC)	Upper girth weld (UGW), lower flange weld (LFW), upper axial welds (UAW), and lower support forging or casting	Enhanced visual (EVT-1) examination, no later than 2 refueling outages from the beginning of the license renewal period and subsequent examination on a ten-year interval.	A minimum of 25% of one side of the circumference of the surface of the UFW and adjacent base metal shall be examined. (Notes 8 and 10) See Figure 4-13.
MRP-227 Revision 1-A entries **W3.Core Barrel Assembly** Upper flange Weld (UFW)	All plants	Cracking (SCC)	W3.1.Upper girth weld (UGW), W3.3.lower flange weld (LFW), W3.2.upper axial welds (UAW), and W3.4.lower support forging or casting	Enhanced visual (EVT-1) examination, no later than 2 refueling outages from the beginning of the license renewal period and subsequent examination on a ten-year interval.	100% of the accessible weld length of one side of the UFW and ¾" of adjacent base metal shall be examined. (Note 6) See Figure 4-13.

TABLE 4-3: Westinghouse plants Primary Items

Item	Applicability	Effect (Mechanism)	Expansion Link	Exam. Meth./ Freq.	Exam. Coverage
Existing MRP-227-A entries **Core Barrel Assembly** Upper and lower core barrel cylinder girth welds	All plants	Cracking (SCC, IASCC, Fatigue)	Upper and lower core barrel cylinder axial welds	Periodic enhanced visual (EVT-1) examination, no later than 2 refueling outages from the beginning of the license renewal period and subsequent examination on a ten-year interval.	100% of one side of the accessible surfaces of the selected weld and adjacent base metal (Note 4). See Figure 4-22
MRP-227 Revision 1 entries **W4.Core Barrel Assembly** Lower girth weld (LGW) *ALSO– Upper girth weld (UGW) and Lower flange weld (LFW) have been changed to expansion items Table 4-6*	All plants	Cracking (SCC, IASCC, Fatigue) Aging Management (IE)	Upper core plate, Lower support column bodies (cast, non-cast), middle axial welds (MAW), lower axial welds (LAW)	Enhanced visual (EVT-1) examination, no later than 2 refueling outages from the beginning of the license renewal period and subsequent examination on a ten-year interval.	A minimum of 25% of the OD circumference of the LGW and adjacent base metal shall be examined. (Notes 8 and 10) See Figure 4-13.

TABLE 4-3: Westinghouse plants Primary Items

Item	Applicability	Effect (Mechanism)	Expansion Link	Exam. Meth./ Freq.	Exam. Coverage
MRP-227 Revision 1-A entries **W4.Core Barrel Assembly** Lower girth weld (LGW)	All plants	Cracking (SCC, IASCC) Aging Management (IE)	W4.1.Upper core plate, W4.4.Lower support column bodies (cast, non-cast), W4.2.middle axial welds (MAW), W4.3.lower axial welds (LAW)	Enhanced visual (EVT-1) examination, no later than 2 refueling outages from the beginning of the license renewal period and subsequent examination on a ten-year interval.	100% of the accessible weld length of the OD of the LGW and ¾" of adjacent base metal shall be examined. (Note 6) See Figure 4-13.
Existing MRP-227-A entries **Baffle-Former Assembly** Baffle-edge bolts	All plants with baffle-edge bolts	Cracking (IASCC, Fatigue) that results in • Lost or broken locking devices • Failed or missing bolts • Protrusion of bolt heads Aging Management (IE and ISR) (Note 6)	None	Visual (VT-3) examination, with baseline examination between 20 and 40 EFPY and subsequent examinations on a ten-year interval.	Bolts and locking devices on high fluence seams. 100% of components accessible from core side (Note 3). See Figure 4-23.

TABLE 4-3: Westinghouse plants Primary Items

Item	Applicability	Effect (Mechanism)	Expansion Link	Exam. Meth./ Freq.	Exam. Coverage
MRP-227 Revision 1 entries **W5.Baffle-Former Assembly** Baffle-edge bolts	All plants with baffle-edge bolts	Cracking (IASCC, Fatigue) that results in • Lost or broken locking devices • Failed or missing bolts • Protrusion of bolt heads Aging Management (IE and ISR) (Note 6)	None	Visual (VT-3) examination, with baseline examination between 20 and 40 EFPY and subsequent examinations on a ten-year interval.	Bolts and locking devices on high fluence seams. 100% of components accessible from core side. See Figure 4-14.
MRP-227 Revision 1-A entries **W5.Baffle-Former Assembly** Baffle-edge bolts	Entry remained unchanged within MRP-227 Revision 1-A.				
Existing MRP-227-A entries **Baffle-Former Assembly** Baffle-former bolts	All plants	Cracking (IASCC, Fatigue) Aging Management (IE and ISR) (Note 6)	Lower support column bolts, Barrel-former bolts	Baseline volumetric (UT) examination between 25 and 35 EFPY, with subsequent examination on a ten-year interval.	100% of accessible bolts (Note 3). Heads accessible from the core side. UT accessibility may be affected by complexity of head and locking device designs. See Figures 4-23 and 4-24.

TABLE 4-3: Westinghouse plants Primary Items

Item	Applicability	Effect (Mechanism)	Expansion Link	Exam. Meth./ Freq.	Exam. Coverage
MRP-227 Revision 1 entries **W6.Baffle-Former Assembly** Baffle-former bolts (Note 9)	All plants	Cracking (IASCC, Fatigue) Aging Management (IE and ISR) (Note 6)	Lower support column bolts, Barrel-former bolts	Baseline volumetric (UT) examination between 25 and 35 EFPY, with subsequent examination on a ten-year interval.	100% of accessible bolts. (Note 3) See Figure 4-15.
MRP-227 Revision 1-A entries **W6.Baffle-Former Assembly** Baffle-former bolts (Note 7)	All plants	Cracking (IASCC, Fatigue) Aging Management (IE and ISR) (Note 4)	W6.2.Lower support column bolts, W6.1.Barrel-former bolts	Baseline volumetric (UT) examination interval is dependent on the plant design (Note 8). Subsequent examination is dependent on the plant design and the results of the baseline inspection (Note 9).	100% of accessible bolts. (Note 3) See Figure 4-15.

TABLE 4-3: Westinghouse plants Primary Items

Item	Applicability	Effect (Mechanism)	Expansion Link	Exam. Meth./ Freq.	Exam. Coverage
Existing MRP-227-A entries **Baffle-Former Assembly** Assembly (Includes: Baffle plates, baffle edge bolts and indirect effects of void swelling in former plates)	All plants	Distortion (Void Swelling), or Cracking (IASCC) that results in • Abnormal interaction with fuel assemblies • Gaps along high fluence baffle joint • Vertical displacement of baffle plates near high fluence joint • Broken or damaged edge bolt locking systems along high fluence baffle joint	None	Visual (VT-3) examination to check for evidence of distortion, with baseline examination between 20 and 40 EFPY and subsequent examinations on a ten-year interval.	Core side surface as indicated. See Figures 4-24, 4-25, 4-26 and 4-27.
MRP-227 Revision 1 entries **W7.Baffle-Former Assembly** Assembly (Includes: Baffle plates, baffle edge bolts, corner bolts, and indirect effects of void swelling in former plates)	All plants	Distortion (Void Swelling), or Cracking (IASCC) that results in • Abnormal interaction with fuel assemblies • Gaps between plates • Vertical displacement of baffle plates • Broken or damaged edge bolts	None	Visual (VT-3) examination to check for evidence of distortion, with baseline examination between 20 and 40 EFPY and subsequent examinations on a ten-year interval.	Core side surface: • High fluence baffle joints • Top and bottom edge of baffle plates • Bolts and locking devices See Figure 4-16.

TABLE 4-3: Westinghouse plants Primary Items

Item	Applicability	Effect (Mechanism)	Expansion Link	Exam. Meth./ Freq.	Exam. Coverage
MRP-227 Revision 1-A entries **W7.Baffle-Former Assembly** Assembly (Includes: Baffle plates, baffle edge bolts, corner bolts, and indirect effects of void swelling in former plates)	Entry remained unchanged within MRP-227 Revision 1-A.				
Existing MRP-227-A entries **Alignment and Interfacing Components** Internals hold down spring	All plants with 304 stainless steel hold down springs	Distortion (Loss of Load) Note: This mechanism was not strictly identified in the original list of age-related degradation mechanisms [7].	None	Direct measurement of spring height within three cycles of the beginning of the license renewal period. If the first set of measurements is not sufficient to determine life, spring height measurements must be taken during the next two outages, in order to extrapolate the expected spring height to 60 years.	Measurements should be taken at several points around the circumference of the spring, with a statistically adequate number of measurements at each point to minimize uncertainty. See Figure 4-28.
MRP-227 Revision 1 entries **W8.Alignment and Interfacing Components** Internals hold down spring	All plants with 304 stainless steel hold down springs	Distortion (Loss of Load due to Stress Relaxation)	None	Direct measurement of spring height within three cycles of the beginning of (before or after) the license renewal period. If the first set of measurements is not sufficient to assess remaining life, additional spring height measurements will be required.	Measurements should be taken at several points around the circumference of the spring, with a statistically adequate number of measurements at each point to minimize uncertainty. See Figure 4-17.

TABLE 4-3: Westinghouse plants Primary Items

Item	Applicability	Effect (Mechanism)	Expansion Link	Exam. Meth./ Freq.	Exam. Coverage
MRP-227 Revision 1-A entries **W8.Alignment and Interfacing Components** Internals hold down spring	Entry remained unchanged within MRP-227 Revision 1-A.				
Existing MRP-227-A entries **Thermal Shield Assembly** Thermal shield flexures	All plants with thermal shields	Cracking (Fatigue) or Loss of Material (Wear) that results in thermal shield flexures excessive wear, fracture, or complete separation	None	Visual (VT-3) no later than 2 refueling outages from the beginning of the license renewal period. Subsequent examinations on a ten-year interval.	100% of thermal shield flexures. See Figures 4-29 and 4-36.
MRP-227 Revision 1 entries **W9.Thermal Shield Assembly** Thermal shield flexures	All plants with thermal shields	Cracking (Fatigue) or Loss of Material (Wear) that results in thermal shield flexures excessive wear, fracture, or complete separation	None	Visual (VT-3) no later than 2 refueling outages from the beginning of the license renewal period. Subsequent examinations on a ten-year interval.	100% of thermal shield flexures. See Figure 4-18.
MRP-227 Revision 1-A entries **W9.Thermal Shield Assembly** Thermal shield flexures	Entry remained unchanged within MRP-227 Revision 1-A.				

TABLE 4-3: Westinghouse plants Primary Items

Existing MRP-227-A entries

Notes to Table 4-3:

1. Examination acceptance criteria and expansion criteria for the Westinghouse components are in Table 5-3.
2. A minimum of 75% of the total identified sample population must be examined.
3. A minimum of 75% of the total population (examined + unexamined), including coverage consistent with the Expansion criteria in Table 5-3, must be examined for inspection credit.
4. A minimum of 75% of the total weld length (examined + unexamined), including coverage consistent with the Expansion criteria in Table 5-3, must be examined from either the inner or outer diameter for inspection credit.
5. The lower core barrel flange weld may be alternatively designated as the core barrel-to-support plate weld in some Westinghouse plant designs.
6. Void swelling effects on this component is managed through management of void swelling on the entire baffle-former assembly.

MRP-227 Revision 1 entries

Notes to Table 4-3:

1. Examination acceptance criteria and expansion criteria for the Westinghouse components are in Table 5-3.
2. A minimum of 75% of the total identified sample population must be examined.
3. A minimum of 75% of the total bolt population (examined + unexamined), including coverage consistent with the Expansion criteria in Table 5-3, must be examined for inspection credit.
4. A minimum of 25% of the total weld length (examined + unexamined), including coverage consistent with the Expansion criteria in Table 5-3, must be examined from either the inner or outer diameter for inspection credit.
5. The lower core barrel flange weld may be alternatively designated as the core barrel-to-support plate weld in some Westinghouse plant designs.
6. Void swelling effects on this component are managed through management of void swelling on the entire baffle-former assembly.
7. In WCAP-17451-P Revision 1 the baseline examination schedule has been adjusted for various CRGT designs, the extent of individual CRGT examination modified, and flexible subsequent examination regimens correlating to initial baseline sample size, accuracy of wear estimation and examination results. Initial inspection prior to the license renewal period may be required.
8. Examination coverage requires 25% of the circumference of either the ID or the OD of the weld.
9. Baffle-former bolt inspection includes inspection of the corner plate bolts when applicable.
10. The stated coverage requirement is the minimum if no significant indications are found. However the Examination Acceptance criteria in Section 5 require that additional coverage must be achieved in the same outage if significant flaws are found. This contingency should be considered for inspection planning purposes.

TABLE 4-3: Westinghouse plants Primary Items

MRP-227 Revision 1-A entries

Notes to Table 4-3:

1. Examination acceptance criteria and expansion criteria for the Westinghouse components are in Table 5-3.
2. A minimum of 75% of the total identified sample population must be examined.
3. A minimum of 75% of the total bolt population (examined + unexamined), including coverage consistent with the Expansion criteria in Table 5-3, must be examined for inspection credit.
4. Void swelling effects on this component are managed through management of void swelling on the entire baffle-former assembly.
5. In WCAP-17451-P the baseline examination schedule has been adjusted for various CRGT designs, the extent of individual CRGT examination modified, and flexible subsequent examination regimens correlating to initial baseline sample size, accuracy of wear estimation and examination results. Initial inspection prior to the license renewal period may be required. Use WCAP-17451-P [37], Revision 1, including the modified requirements due to the interim guidance provided in letter OG-18-46 [38].
6. Examination coverage requires a minimum of 50% of the length of either the ID or the OD of the weld being examined.
7. Baffle-former bolt inspection includes inspection of the corner plate bolts when applicable.
8. In accordance with MRP 2017-009 [39] and MRP 2017-010 [41], Tier 1 plants are to perform the baseline UT examination by 20 EFPY or during the next refueling outage after March 1, 2016. Per MRP 2017-009 [39], Tier 2 plants are to perform the baseline UT examination at no later than 30 EFPY (initial Tier 2 plant baseline UT exams performed prior to 1/1/2018 are acceptable). All other remaining plants are to perform the baseline UT examination at no later than 35 EFPY.
9. Re-examination periods shall be determined by plant-specific evaluation per the MRP-227 Needed Requirement 7.5 as documented and dispositioned in the owner's plant corrective action program. If atypical or aggressive baffle-former bolt degradation as defined in MRP 2017-009 [39] (i.e., ≥3% of baffle-former bolts with UT or visual indications or clustering* for downflow plants and ≥5% of baffle-former bolts with UT or visual indications or clustering* for upflow plants) is observed, the interim guidance (MRP 2016-021 [40] and MRP 2017-009 [39]) provides limitations to the permitted reinspection interval (not to exceed 6 years maximum) unless further evaluation is performed to justify a longer interval (See Applicant/Licensee Action Item 1 in the NRC SE for evaluation submittal requirements [35]). If evaluation justifies a longer reinspection interval, it is not permitted to exceed 10 years.

*"Clustering" is defined per NSAL-16-1 Rev. 1 [42] as three or more adjacent defective baffle-former bolts or more than 40% defective baffle-former bolts on the same baffle plate. Untestable bolts should be reviewed on a plant-specific basis consistent with WCAP-17096-NP-A for determination if these should be considered when evaluating clustering.

TABLE 4-4: B&W plants Expansion Items

Item	Applicability	Effect (Mechanism)	Primary Link	Exam. Meth./ Freq.	Exam. Coverage
Existing MRP-227-A entries **Upper Grid Assembly** Alloy X-750 dowel-to-upper grid fuel assembly support pad welds	All plants (except DB)	Cracking (SCC), including the detection of separated or missing locking welds, or missing dowels	Alloy X-750 dowel-to-guide block welds	Visual (VT-3) examination. Subsequent examinations on the 10-year ISI interval unless an applicant/licensee provides an evaluation for NRC staff approval that justifies a longer interval between inspections.	Accessible surfaces of 100% of the dowel locking welds. See Figure 4-6 (i.e., these are similar to the lower grid fuel assembly support pads).
MRP-227 Revision 1 entries **B13.1.Upper Grid Assembly** Alloy X-750 dowel-to-upper grid fuel assembly support pad welds	All plants (Note 2)	Cracking (SCC), including the detection of separated or missing locking welds, or missing dowels	B13.Alloy X-750 dowel-to-guide block welds	Visual (VT-3) examination. Subsequent examination during each 10-year interval unless an applicant/licensee provides an evaluation for NRC staff approval that justifies a longer interval between inspections.	Accessible surfaces of 100% of the support pad dowel locking welds. See Figure 4-6. (These are similar to the lower grid fuel assembly support pads)
MRP-227 Revision 1-A entries **B13.1.Upper Grid Assembly** Alloy X-750 dowel-to-upper grid fuel assembly support pad welds	Entry remained unchanged within MRP-227 Revision 1-A.				
Existing MRP-227-A entries No entry exists in MRP-227-A for this item **Vent Valve Assembly** Vent valve bodies	-	-	-	-	-

TABLE 4-4: B&W plants Expansion Items

Item	Applicability	Effect (Mechanism)	Primary Link	Exam. Meth./ Freq.	Exam. Coverage
MRP-227 Revision 1 entries					
B2.1.Vent Valve Assembly Vent valve bodies	All plants	Cracking (TE), including the detection of fractured vent valve bodies, surface irregularities (damaged, grossly cracked, or missing portions) of the vent valve bodies, as a result of damage to the jackscrews or locking devices (Note 3)	B2.CRGT spacer castings	Visual (VT-3) examination. Subsequent examination during each 10-year interval unless an applicant/licensee provides an evaluation for NRC staff approval that justifies a longer interval between inspections.	Accessible surfaces of 100% of the vent valve bodies on the plenum side (core side) of the vent valve. (Limited accessibility) See Figures 4-9 and 4-10. (Note 4)
MRP-227 Revision 1-A entries					
B2.1.Vent Valve Assembly Vent valve bodies (Note 9)	NO CHANGE	NO CHANGE	NO CHANGE	NO CHANGE	NO CHANGE

TABLE 4-4: B&W plants Expansion Items

Item	Applicability	Effect (Mechanism)	Primary Link	Exam. Meth./ Freq.	Exam. Coverage
Existing MRP-227-A entries **Core Barrel Assembly** Upper thermal shield (UTS) bolts and their locking devices	All plants	Bolt or Stud/Nut: Cracking (SCC) Locking Devices: Loss of material, damaged, distorted or missing locking devices (Wear or Fatigue damage by failed bolts).	UCB, LCB or FD bolts and their locking devices	Bolt or Stud/Nut: Volumetric examination (UT). Locking Devices: Visual (VT-3) Examination Subsequent examinations on the 10-year ISI interval unless an applicant/licensee provides an evaluation for NRC staff approval that justifies a longer interval between inspections.	100% of accessible bolts or studs/nuts and their locking devices (Note 2). See Figure 4-7.
Core Barrel Assembly Surveillance specimen holder tube (SSHT) studs/nuts (CR-3) or bolts (DB) and their locking devices	CR-3, DB				
MRP-227 Revision 1 entries **B7.1.Core Barrel Assembly** Upper thermal shield bolts (UTS) and their locking devices	All plants	Bolts: Cracking (SCC—UTS and SSHT bolts, IC/ISR/Wear/Fatigue–SSHT bolts) (Note 7) Locking Devices: Loss of material, damaged, distorted or missing locking devices (Wear or Fatigue damage by failed bolts)	B7, B8, B12, UCB, LCB, or FD bolts and their locking devices B14.Shock pad bolts and their locking devices (TMI-1 only)	Bolts: Volumetric (UT) examination. Locking Devices: Visual (VT-3) examination. Subsequent examination during each 10-year interval unless an applicant/licensee provides an evaluation for NRC staff approval that justifies a longer interval between inspections.	100% of accessible bolts and their locking devices. (Note 5) See Figure 4-7 (for UTS bolts) and Figure 4-8(b) (for SSHT bolts).
B7.2.Core Barrel Assembly Surveillance specimen holder tube (SSHT) bolts and their locking devices	DB				

TABLE 4-4: B&W plants Expansion Items

Item	Applicability	Effect (Mechanism)	Primary Link	Exam. Meth./ Freq.	Exam. Coverage
MRP-227 Revision 1-A entries **B7.1.Core Barrel Assembly** Upper thermal shield (UTS) bolts and their locking devices	NO CHANGE	NO CHANGE	B7.UCB, B8.LCB, or B12.FD bolts and their locking devices B14.Shock pad bolts and their locking devices (TMI-1 only)	NO CHANGE	NO CHANGE
B7.2.Core Barrel Assembly Surveillance specimen holder tube (SSHT) bolts and their locking devices	NO CHANGE				
Existing MRP-227-A entries **Core Barrel Assembly** Core barrel cylinder (including vertical and circumferential seam welds) Former plates	All plants	Cracking (IE), including readily detectable cracking	Baffle plates	No examination requirements. Justify by evaluation or by replacement.	Inaccessible. See Figure 4-2.
MRP-227 Revision 1 entries **Core Barrel Assembly** B10.1.Core barrel cylinder (including vertical and center circumferential seam welds) B10.2.Former plates	All plants	Cracking (IE) (Note 3)	B10.Baffle plates	No examination requirements. Justify by evaluation or by replacement.	Inaccessible. See Figure 4-2.
MRP-227 Revision 1-A entries B10.1.Core barrel cylinder (including vertical and center circumferential seam welds) B10.2.Former plates	Entry remained unchanged within MRP-227 Revision 1-A.				

TABLE 4-4: B&W plants Expansion Items

Item	Applicability	Effect (Mechanism)	Primary Link	Exam. Meth./ Freq.	Exam. Coverage
Existing MRP-227-A entries **Core Barrel Assembly** Baffle-to-baffle bolts Core barrel-to-former bolts	All plants	Cracking (IASCC, IE, Overload) (Note 3)	Baffle-to-former bolts	Internal baffle-to-baffle bolts: No examination requirements, Justify by evaluation or by replacement.	An acceptable examination technique currently not available. See Figure 4-2.
				External baffle-to-baffle bolts, core barrel-to-former bolts: No examination requirements. Justify by evaluation or by replacement.	Inaccessible. See Figure 4-2.
MRP-227 Revision 1 entries **Core Barrel Assembly** B9.1.Baffle-to-baffle bolts B9.2.Core barrel-to-former bolts	All plants	Cracking (IASCC, IE, IC/ISR/Fatigue/Wear, Overload) (Notes 3, 6, 8)	B9.Baffle-to-former bolts	Internal baffle-to-baffle bolts: No examination requirements. Justify by evaluation or by replacement.	An acceptable examination technique currently not available. See Figure 4-2.
				External baffle-to-baffle bolts, Core barrel-to-former bolts: No examination requirements. Justify by evaluation or by replacement.	Inaccessible. See Figure 4-2.
MRP-227 Revision 1-A entries B9.1.Baffle-to-baffle bolts B9.2.Core barrel-to-former bolts	Entry remained unchanged within MRP-227 Revision 1-A.				

TABLE 4-4: B&W plants Expansion Items

Item	Applicability	Effect (Mechanism)	Primary Link	Exam. Meth./ Freq.	Exam. Coverage
<u>Existing MRP-227-A entries</u> **Core Barrel Assembly** Locking devices, including locking welds, for the external baffle-to-baffle bolts and core barrel-to-former bolts	All plants	Cracking (IASCC, IE)	Locking devices, including locking welds, of baffle-to-former bolts or internal baffle-to-baffle bolts	No examination requirements. Justify by evaluation or by replacement.	Inaccessible. See Figure 4-2.
<u>MRP-227 Revision 1 entries</u> **Core Barrel Assembly** B11.1.Locking devices, including locking welds, of the external baffle-to-baffle bolts and core barrel-to-former bolts	All plants	Cracking (IE)	B11.Locking devices, including locking welds, of baffle-to-former bolts and internal baffle-to-baffle bolts	No examination requirements. Justify by evaluation or by replacement.	Inaccessible. See Figure 4-2.
<u>MRP-227 Revision 1-A entries</u> **Core Barrel Assembly** B11.1.Locking devices, including locking welds, of the external baffle-to-baffle bolts and core barrel-to-former bolts	Entry remained unchanged within MRP-227 Revision 1-A.				

TABLE 4-4: B&W plants Expansion Items

Item	Applicability	Effect (Mechanism)	Primary Link	Exam. Meth./ Freq.	Exam. Coverage
Existing MRP-227-A entries **Lower Grid Assembly** Lower grid fuel assembly support pad items: pad, pad-to-rib section welds, Alloy X-750 dowel, cap screw, and their locking welds (Note: the pads, dowels and cap screws are included because of IE of the welds)	All plants	Cracking (IE), including the detection of separated or missing welds, missing support pads, dowels, cap screws and locking welds, or misalignment of the support pads	IMI guide tube spiders and spider-to-lower grid rib section welds	Visual (VT-3) examination. Subsequent examinations on the 10-year ISI interval unless an applicant/licensee provides an evaluation for NRC staff approval that justifies a longer interval between inspections.	Accessible surfaces of the pads, dowels, and cap screws, and associated welds in 100% of the lower grid fuel assembly support pads. See Figure 4-6.
MRP-227 Revision 1 entries **B15.1.Lower Grid Assembly** Lower grid fuel assembly support pad component items: pad, pad-to-rib section welds, Alloy X-750 dowel, cap screws, and their locking welds (Note: the pads, dowels, and cap screws are included because of IE of the welds)	All plants	Cracking (IE), including the detection of separated or missing welds, missing support pads, dowels, cap screws and locking welds, or misalignment of the support pads (Note 3)	B15.IMI guide tube spiders and spider-to-lower grid rib section welds	Visual (VT-3) examination. Subsequent examination during each 10-year interval unless an applicant/licensee provides an evaluation for NRC staff approval that justifies a longer interval between inspections.	Accessible surfaces of the pads, dowels, cap screws, and associated welds for 100% of the lower grid fuel assembly support pads. See Figure 4-6.

TABLE 4-4: B&W plants Expansion Items

Item	Applicability	Effect (Mechanism)	Primary Link	Exam. Meth./ Freq.	Exam. Coverage
MRP-227 Revision 1-A entries **B15.1.Lower Grid Assembly** Lower grid fuel assembly support pad component items: pad, pad-to-rib section welds, Alloy X-750 dowel, cap screws, and their locking welds (Note: the pads, dowels, and cap screws are included because of IE of the welds)	Entry remained unchanged within MRP-227 Revision 1-A.				
Existing MRP-227-A entries **Lower Grid Assembly** Alloy X-750 dowel-to-lower grid fuel assembly support pad welds	All plants	Cracking (SCC), including the detection of separated or missing locking welds, or missing dowels	Alloy X-750 dowel-to-guide block welds	Visual (VT-3) examination. Subsequent examinations on the 10-year ISI interval unless an applicant/licensee provides an evaluation for NRC staff approval that justifies a longer interval between inspections.	Accessible surfaces of 100% of the support pad dowel locking welds. See Figure 4-6.
MRP-227 Revision 1 entries **B13.2.Lower Grid Assembly** Alloy X-750 dowel-to-lower grid fuel assembly support pad welds	All plants (Note 2)	Cracking (SCC), including the detection of separated or missing locking welds, or missing dowels	B13.Alloy X-750 dowel-to-guide block welds	Visual (VT-3) examination. Subsequent examination during each 10-year interval unless an applicant/licensee provides an evaluation for NRC staff approval that justifies a longer interval between inspections.	Accessible surfaces of 100% of the support pad dowel locking welds. See Figure 4-6.

TABLE 4-4: B&W plants Expansion Items

Item	Applicability	Effect (Mechanism)	Primary Link	Exam. Meth./ Freq.	Exam. Coverage
MRP-227 Revision 1-A entries **B13.2.Lower Grid Assembly** Alloy X-750 dowel-to-lower grid fuel assembly support pad welds	Entry remained unchanged within MRP-227 Revision 1-A.				
Existing MRP-227-A entries **Lower Grid Assembly** Lower grid shock pad bolts and their locking devices	TMI-1	Bolts: Cracking (SCC) Locking Devices: Loss of material, damaged, distorted or missing locking devices (Wear or Fatigue damage by failed bolts).	UCB, LCB or FD bolts and their locking devices	Bolt: Volumetric examination (UT). Locking Devices: Visual (VT-3) examination. Subsequent examinations on the 10-year ISI interval unless an applicant/licensee provides an evaluation for NRC staff approval that justifies a longer interval between inspections.	100% of accessible bolts and their locking devices. (Note 2) See Figure 4-4.

TABLE 4-4: B&W plants Expansion Items

Item	Applicability	Effect (Mechanism)	Primary Link	Exam. Meth./ Freq.	Exam. Coverage
<u>MRP-227 Revision 1 entries</u> **This item is now a *<u>Primary</u>* item in Table 4-1:** **B14.Lower Grid Assembly** Shock pad bolts and their locking devices	TMI-1	Bolts: Cracking (SCC, Fatigue) Locking Devices: Loss of material, damaged or distorted or missing locking devices (Wear or Fatigue damage by failed bolts)	*<u>Expansion link is</u>*: UTS bolts and LTS bolts and their locking devices	Bolts: Volumetric (UT) examination during the next 10-year ISI interval. Locking Devices: Visual (VT-3) examination during the next 10-year ISI interval. Subsequent examination during each 10-year ISI interval unless an evaluation of the baseline results, submitted for NRC staff approval, justifies a longer interval between examinations..	100% of accessible bolts and their locking devices. (Note 6) See Figure 4-4.
<u>MRP-227 Revision 1-A entries</u> **This item is now a *<u>Primary</u>* item in Table 4-1:** **B14.Lower Grid Assembly** Shock pad bolts and their locking devices	Entry remained unchanged within MRP-227 Revision 1-A.				

TABLE 4-4: B&W plants Expansion Items

Item	Applicability	Effect (Mechanism)	Primary Link	Exam. Meth./ Freq.	Exam. Coverage
Existing MRP-227-A entries **Lower Grid Assembly** Lower thermal shield (LTS) bolts (ANO-1, DB and TMI-1) or studs/nuts (ONS, CR-3) and their locking devices	All plants	Bolts or Studs/Nuts: Cracking (SCC) Locking Devices: Loss of material, damaged, distorted or missing locking devices (Wear or Fatigue damage by failed bolts).	UCB,LCB or FD bolts and their locking devices	Bolt or Stud/Nut: Volumetric examination (UT). Locking Devices: Visual (VT-3) examination. Subsequent examinations on the 10-year ISI interval unless an applicant/licensee provides an evaluation for NRC staff approval that justifies a longer interval between inspections.	100% of accessible bolts and their locking devices. (Note 2) See Figure 4-8.
MRP-227 Revision 1 entries **B8.1.Lower Grid Assembly** Lower thermal shield (LTS) bolts (ANO-1, DB, and TMI-1) or studs/nuts (ONS-1, ONS-2, and ONS-3) and their locking devices	All plants	Bolts or Studs/Nuts: Cracking (SCC) Locking Devices: Loss of material, damaged, distorted or missing locking devices (Wear or Fatigue damage by failed bolts or studs/nuts)	B7, B8, B12. UCB, LCB, or FD bolts and their locking devices B14. Shock pad bolts and their locking devices (TMI-1 only)	Bolts or Studs/Nuts: Volumetric (UT) examination. Locking Devices: Visual (VT-3) examination. Subsequent examination during each 10-year interval unless an applicant/licensee provides an evaluation for NRC staff approval that justifies a longer interval between inspections.	100% of accessible bolts or studs/nuts and their locking devices. (Note 5) See Figure 4-8.

TABLE 4-4: B&W plants Expansion Items

Item	Applicability	Effect (Mechanism)	Primary Link	Exam. Meth./ Freq.	Exam. Coverage
MRP-227 Revision 1-A entries **B8.1.Lower Grid Assembly** Lower thermal shield (LTS) bolts (ANO-1, DB, and TMI-1) or studs/nuts (ONS-1, ONS-2, and ONS-3) and their locking devices	NO CHANGE	NO CHANGE	B7.UCB, B8.LCB, or B12.FD bolts and their locking devices B14. Shock pad bolts and their locking devices (TMI-1 only)	NO CHANGE	NO CHANGE
Entry not Existing in MRP-227-A *The following item did not exist in the MRP-227-A table entries.*					
MRP-227-Revision 1 entries **Lower Grid Assembly** B10.3.Lower grid rib section	All plants	Cracking (IE), including the detection of readily detectable cracking (Note 3)	B10.Baffle plates	Visual (VT-3) examination. Subsequent examination during each 10-year interval unless an applicant/licensee provides an evaluation for NRC staff approval that justifies a longer interval between inspections.	100% of accessible surfaces of the lower grid rib section heat-affected zone (HAZ) adjacent to the spider-to-lower grid rib section welds. See Figures 4-3 and 4-6.
MRP-227-Revision 1-A entries B10.3.Lower grid rib section	Entry remained unchanged within MRP-227 Revision 1-A.				

TABLE 4-4: B&W plants Expansion Items

Item	Applicability	Effect (Mechanism)	Primary Link	Exam. Meth./ Freq.	Exam. Coverage

Existing MRP-227-A entries

Notes to Table 4-4:

1. Examination acceptance criteria and expansion criteria for the B&W components are in Table 5-1.
2. A minimum of 75% of the total population (examined + unexamined) must be examined for inspection credit.
3. The primary aging degradation mechanisms for loss of joint tightness for these items are IC and ISR. Fatigue and Wear, which can also lead to cracking, are secondary aging degradation mechanisms after significant stress relaxation and loss of preload has occurred due to IC/ISR. Bolt stress relaxation cannot readily be inspected by NDE. Only bolt cracking could be inspected by UT inspection if it were possible for these bolts. Therefore, the effects of loss of joint tightness and/or cracking on the functionality of these bolts relative to the entire core barrel assembly will be addressed by analysis of the core barrel assembly, which will be performed to address Applicant/Licensee action item 6 of the SE [27].

MRP-227 Revision 1 entries

Notes to Table 4-4:

1. Examination acceptance criteria and expansion criteria for the B&W component items and welds are in Sections 5.1 - 5.3 of this document.
2. For the alternate configuration at DB, the difference in configuration shall be addressed on a unit-specific basis.
3. Loss of ductility and fracture toughness, which are induced by thermal aging or neutron irradiation embrittlement, are not directly monitored. These effects are indirectly managed by performing periodic visual examinations capable of detecting cracking in the component.
4. It is known that some of the vent valves originally-installed to the B&W units were replaced with spares due to locking device issues. While the ferrite contents of the originally-installed vent valve bodies are known, the serial number [S/N] and corresponding ferrite contents of the currently-installed vent valve bodies is not fully known for the B&W units. Therefore, each utility shall identify the body S/N and heat number for each installed vent valve to determine installed heat of material and the ferrite content. The S/N numbers may be visible with an underwater video camera, e.g., during vent valve exercising. If the ferrite content of the currently-installed vent valve bodies can be verified to be under the TE screening criterion for the material, the vent valve bodies can be removed as an Expansion component item for the particular B&W unit and become a No Additional Measures component item. However, since vent valve assemblies are typically replaced in whole rather than by part, each time any changes are made to vent valve assemblies (e.g., replacement of a whole assembly), the vent valve body serial number, heat number, and ferrite content need to be verified. If, at any time, the ferrite content of any installed vent valve body is greater than the screening criterion for thermal embrittlement, that vent valve body will need to be included as an Expansion component item.
5. A minimum of 75% of the total population (examined + unexamined) must be examined for inspection credit.
6. The primary aging degradation mechanisms for loss of joint tightness for this item are IC and ISR. Fatigue and wear, which can also lead to cracking, are secondary aging degradation mechanisms after significant stress relaxation and loss of preload has occurred due to IC/ISR. Bolt stress relaxation cannot readily be inspected by NDE. Only bolt cracking could be inspected by UT inspection if it were possible for these bolts. Therefore, the effects of loss of joint tightness and/or cracking on the functionality of these bolts relative to the entire core barrel assembly will be addressed by analysis of the core barrel assembly.
7. This table entry for the SSHT bolts also includes the aging degradation mechanisms of ISR/IC/wear/fatigue for the compression collar and washer of the SSHT bolt.
8. The aging degradation mechanism consequence of overload is only applicable to the internal baffle-to-baffle bolts, not the core barrel-to-former bolts and external baffle-to-baffle bolts.

TABLE 4-4: B&W plants Expansion Items

Item	Applicability	Effect (Mechanism)	Primary Link	Exam. Meth./ Freq.	Exam. Coverage

MRP-227 Revision 1-A entries

Notes to Table 4-4:

1. Examination acceptance criteria and expansion criteria for the B&W component items and welds are in Sections 5.1 - 5.3 of this document. Degradation exhibited in any of these primary links requires expansion in accordance with the criteria noted in Table 5-1.
2. For the alternate configuration at DB, the difference in configuration shall be addressed on a unit-specific basis.
3. Loss of ductility and fracture toughness, which are induced by thermal aging or neutron irradiation embrittlement, are not directly monitored. These effects are indirectly managed by performing periodic visual examinations capable of detecting cracking in the component.
4. It is known that some of the vent valves originally-installed to the B&W units were replaced with spares due to locking device issues. While the ferrite contents of the originally-installed vent valve bodies are known, the serial number [S/N] and corresponding ferrite contents of the currently-installed vent valve bodies is not fully known for the B&W units. Therefore, each utility shall identify the body S/N and heat number for each installed vent valve to determine installed heat of material and the ferrite content. The S/N numbers may be visible with an underwater video camera, e.g., during vent valve exercising. If the ferrite content of the currently-installed vent valve bodies can be verified to be under the TE screening criterion for the material, the vent valve bodies can be removed as an Expansion component item for the particular B&W unit and become a No Additional Measures component item. However, since vent valve assemblies are typically replaced in whole rather than by part, each time any changes are made to vent valve assemblies (e.g., replacement of a whole assembly), the vent valve body serial number, heat number, and ferrite content need to be verified. If, at any time, the ferrite content of any installed vent valve body is greater than the screening criterion for thermal embrittlement, that vent valve body will need to be included as an Expansion component item.
5. A minimum of 75% of the total population (examined + unexamined) must be examined for inspection credit.
6. The primary aging degradation mechanisms for loss of joint tightness for this item are IC and ISR. Fatigue and wear, which can also lead to cracking, are secondary aging degradation mechanisms after significant stress relaxation and loss of preload has occurred due to IC/ISR. Bolt stress relaxation cannot readily be inspected by NDE. Only bolt cracking could be inspected by UT inspection if it were possible for these bolts. Therefore, the effects of loss of joint tightness and/or cracking on the functionality of these bolts relative to the entire core barrel assembly will be addressed by analysis of the core barrel assembly.
7. This table entry for the SSHT bolts also includes the aging degradation mechanisms of ISR/IC/wear/fatigue for the compression collar and washer of the SSHT bolt.
8. The aging degradation mechanism consequence of overload is only applicable to the internal baffle-to-baffle bolts, not the core barrel-to-former bolts and external baffle-to-baffle bolts.
9. As noted in Section 3.1.3.1 of the SE for MRP-227, Rev. 1, licensees have the option of screening out the vent valve bodies using certified material test report (CMTR) data, or using the generic technical report PWROG-15032-NP [34] to show that the vent valve bodies have a high probability of having ferrite below the screening criterion for TE.

TABLE 4-5: CE plants Expansion Items

Item	Applicability	Effect (Mechanism)	Primary Link	Exam. Meth./ Freq.	Exam. Coverage
Existing MRP-227-A entries **Core Shroud Assembly (Bolted)** Barrel-shroud bolts	Bolted plant designs	Cracking (IASCC, Fatigue) Aging Management (IE and ISR)	Core shroud bolts	Volumetric (UT) examination. Re-inspection every 10 years following initial inspection.	100% (or as supported by plant-specific justification; Note 2) of barrel-shroud and guide lug insert bolts with neutron fluence exposures > 3 displacements per atom (dpa). See Figure 4-23.
MRP-227 Revision 1 entries **Core Shroud Assembly (Bolted)** C1.2.Barrel-shroud bolts	Bolted plant designs	Cracking (IASCC, Fatigue) Aging Management (IE and ISR)	C1.Core shroud bolts	Volumetric (UT) examination. Re-inspection every 10 years following initial inspection.	100% of accessible barrel shroud bolts or as supported by plant specific justification. See Figure 4-35.
MRP-227 Revision 1-A entries **Core Shroud Assembly (Bolted)** C1.2.Barrel-shroud bolts	Entry remained unchanged within MRP-227 Revision 1-A.				
Existing MRP-227-A entries **Core Support Barrel Assembly** Lower core barrel flange	All plants	Cracking (SCC, Fatigue)	Upper (core support barrel) flange weld	Enhanced visual (EVT-1) examination. Re-inspection every 10 years following initial inspection.	100% of accessible welds and adjacent base metal (Note 2). See Figure 4-15.

TABLE 4-5: CE plants Expansion Items

Item	Applicability	Effect (Mechanism)	Primary Link	Exam. Meth./ Freq.	Exam. Coverage
MRP-227 Revision 1 entries **Core Support Barrel Assembly** C5.1.Lower Girth Weld (LGW)	All plants	Cracking (SCC, Fatigue)	C5.Upper Flange Weld (UFW)	Enhanced visual (EVT-1) examination. Re-inspection every 10 years following initial inspection.	A minimum of 75% of the OD circumferential of the LGW and adjacent base metal shall be examined. See Figure 4-29.
MRP-227 Revision 1-A entries **Core Support Barrel Assembly** C5.1.Lower Girth Weld (LGW)	All plants	Cracking (SCC, Fatigue)	C5.Upper Flange Weld (UFW)	Enhanced visual (EVT-1) examination. Reinspection every 10 years following initial inspection.	100% of the accessible weld length of the OD of the LGW and ¾" of adjacent base metal shall be examined. (Note 5) See Figures 4-29 and 4-30.
Existing MRP-227-A entries **Core Support Barrel Assembly** Upper cylinder (including welds)	All plants	Cracking (SCC) Aging Management (IE)	Upper (core support barrel) flange weld	Enhanced visual (EVT-1) examination. Re-inspection every 10 years following initial inspection.	100% of accessible surfaces of the welds and adjacent base metal (Note 2). See Figure 4-15.
MRP-227 Revision 1 entries **Core Support Barrel Assembly** C5.2.Upper Girth Weld (UGW) and C5.3.Upper Axial Weld (UAW)	All plants	Cracking (SCC) Aging Management (IE)	C5.Upper Flange Weld (UFW)	Enhanced visual (EVT-1) examination. Re-inspection every 10 years following initial inspection.	A minimum of 75% of the OD circumference of the UGW and UAW and adjacent base metal shall be examined. See Figure 4-29.

TABLE 4-5: CE plants Expansion Items

Item	Applicability	Effect (Mechanism)	Primary Link	Exam. Meth./ Freq.	Exam. Coverage
MRP-227 Revision 1-A entries **Core Support Barrel Assembly** C5.2.Upper Girth Weld (UGW) and C5.3.Upper Axial Weld (UAW)	All plants	Cracking (SCC) Aging Management (IE)	C5.Upper Flange Weld (UFW)	Enhanced visual (EVT-1) examination. Reinspection every 10 years following initial inspection.	100% of the accessible weld length of one side of the UGW and UAW and ¾" adjacent base metal shall be examined. (Notes 2 and 5) See Figure 4-29.
Existing MRP-227-A entries **Lower Support Structure** Lower core support beams	All plants except those with core shrouds assembled with full-height shroud plates	Cracking (SCC, Fatigue) including damaged or fractured material	Upper (core support barrel) flange weld	Visual (EVT-1) examination. Re-inspection every 10 years following initial inspection.	100% of accessible surfaces (Note 2). See Figures 4-16 and 4-31.
MRP-227 Revision 1 entries **Lower Support Structure** C5.4.Lower core support beams	All plants except those with welded core shrouds assembled with full-height shroud plates	Cracking (SCC, Fatigue) including damaged or fractured material)	C5.Upper Flange Weld (UFW)	Enhanced visual (EVT-1) examination. Re-inspection every 10 years following initial inspection.	25% of welds and adjacent base metal or locations as justified by plant specific evaluation. See Figure 4-36.
MRP-227 Revision 1-A entries **Lower Support Structure** C5.4.Lower core support beams	All plants except those with welded core shrouds assembled with full-height shroud plates	Cracking (SCC, Fatigue) including damaged or fractured material)	C5.Upper Flange Weld (UFW)	Enhanced visual (EVT-1) examination. Reinspection every 10 years following initial inspection.	25% of welds and ¾" of adjacent base metal or locations as justified by plant-specific evaluation. See Figure 4-35.

TABLE 4-5: CE plants Expansion Items

Item	Applicability	Effect (Mechanism)	Primary Link	Exam. Meth./ Freq.	Exam. Coverage
Existing MRP-227-A entries **Core Support Barrel Assembly** Core barrel assembly axial welds	All plants	Cracking (SCC)	Core barrel assembly girth welds	Enhanced visual (EVT-1) examination, with initial and subsequent examinations dependent on the results of core barrel assembly girth weld examinations.	100% of one side of the accessible weld and adjacent base metal surfaces for the weld with the highest calculated operating stress. See Figure 4-15.
MRP-227 Revision 1 entries **Core Support Barrel Assembly** C6.1.Middle Axial Weld (MAW), C6.2.Lower Axial Weld (LAW)	All plants	Cracking (SCC, IASCC) Aging Management (IE)	C6.Middle Girth Weld (MGW)	Enhanced visual (EVT-1) examination Re-inspection every 10 years following initial inspection.	A minimum of 75% of the OD circumference of the MAW and LAW and adjacent base metal shall be examined. See Figure 4-29.
MRP-227 Revision 1-A entries **Core Support Barrel Assembly** C6.1.Middle Axial Weld (MAW), C6.2.Lower Axial Weld (LAW)	All plants	Cracking (SCC, IASCC) Aging Management (IE)	C6.Middle Girth Weld (MGW)	Enhanced visual (EVT-1) examination Reinspection every 10 years following initial inspection.	100% of the accessible weld length of the OD of the MAW and LAW and ¾" of adjacent base metal shall be examined. (Note 5) See Figure 4-29.

TABLE 4-5: CE plants Expansion Items

Item	Applicability	Effect (Mechanism)	Primary Link	Exam. Meth./ Freq.	Exam. Coverage
Existing MRP-227-A entries **Core Shroud Assembly (Bolted)** Core support column bolts	Bolted plant designs	Cracking (IASCC, Fatigue) Aging Management (IE)	Core shroud bolts	Ultrasonic (UT) examination. Re-inspection every 10 years following initial inspection.	100% (or as supported by plant-specific analysis) of core support column bolts with neutron fluence exposures > 3 dpa (Note 2). See Figures 4-16 and 4-33.
MRP-227 Revision 1 entries **Core Shroud Assembly (Bolted)** C1.1.Core support column bolts	Bolted plant designs	Cracking (IASCC, Fatigue) Aging Management (IE)	C1.Core shroud bolts	Ultrasonic (UT) examination. Re-inspection every 10 years following initial inspection.	100% (or as supported by plant-specific analysis) of core support column bolts (Minimum of 75% of the total population). See Figures 4-37.
MRP-227 Revision 1-A entries **Core Shroud Assembly (Bolted)** C1.1.Core support column bolts	Entry remained unchanged within MRP-227 Revision 1-A.				
Existing MRP-227-A entries **Core Shroud Assembly (Welded)** Remaining axial welds, Ribs and rings (Note: This item has been split into two expansion items C2 and C3 based on core shroud design type and welds)	Plant designs with core shrouds assembled with full-height shroud plates	Cracking (IASCC) Aging Management (IE)	Shroud plates of welded core shroud assemblies	Enhanced visual (EVT-1) examination. Re-inspection every 10 years following initial inspection.	Axial weld seams other than the core shroud reentrant corner welds at the core mid-plane, plus ribs and rings. See Figure 4-13.

TABLE 4-5: CE plants Expansion Items

Item	Applicability	Effect (Mechanism)	Primary Link	Exam. Meth./ Freq.	Exam. Coverage
MRP-227 Revision 1 entries **Core Shroud Assembly (Welded)** C2.1.Remaining axial welds	Plant designs with core shrouds assembled in two vertical sections	Cracking (IASCC), Aging Management (IE)	C2.Core shroud plate-former plate weld	Enhanced visual (EVT-1) examination. Re-inspection every 10 years following initial inspection.	75% of the remaining axial weld length and adjacent base metal as visible from the core side of the shroud other than that already inspected under the primary link. See Figure 4-38.
MRP-227 Revision 1-A entries **Core Shroud Assembly (Welded)** C2.1.Remaining axial welds	Plant designs with core shrouds assembled in two vertical sections	Cracking (IASCC), Aging Management (IE)	C2.Core shroud plate-former plate weld	Enhanced visual (EVT-1) examination. Reinspection every 10 years following initial inspection.	75% of the remaining axial weld length and ¾" of adjacent base metal as visible from the core side of the shroud other than that already inspected under the primary link. See Figure 4-26.
MRP-227 Revision 1 entries **Core Shroud Assembly (Welded)** C3.1.Remaining axial welds, C3.2.Ribs and rings	Only plants with welded core shrouds assembled with full-height shroud plates	Cracking (IASCC), Aging Management (IE)	C3.Shroud plates of welded core shroud assemblies	Enhanced visual (EVT-1) examination. Re-inspection every 10 years following initial inspection.	75% of the remaining axial weld length and adjacent base metal as visible from the core side of the shroud other than that already inspected under the primary link. 25% of the ribs and rings. See Figure 4-38.

TABLE 4-5: CE plants Expansion Items

Item	Applicability	Effect (Mechanism)	Primary Link	Exam. Meth./ Freq.	Exam. Coverage
MRP-227 Revision 1-A entries **Core Shroud Assembly (Welded)** C3.1.Remaining axial welds, C3.2.Ribs and rings	Only plants with welded core shrouds assembled with full-height shroud plates	Cracking (IASCC), Aging Management (IE)	C3.Shroud plates of welded core shroud assemblies	Remaining axial weld: Enhanced visual (EVT-1) examination. Reinspection every 10 years following initial inspection. Ribs and rings: No examination requirements. Justify by evaluation or by replacement.	Remaining axial welds: 75% of the remaining axial weld length and ¾" of adjacent base metal as visible from the core side of the shroud other than that already inspected under the primary link. Ribs and rings: Inaccessible See Figure 4-37.
Existing MRP-227-A entries **Control Element Assembly** Remaining instrument guide tubes	All plants with instrument guide tubes in the CEA shroud assembly	Cracking (SCC, Fatigue) that results in missing supports or separation at the welded joint between the tubes and supports.	Peripheral instrument guide tubes within the CEA shroud assemblies	Visual (VT-3) examination. Re-inspection every 10 years following initial inspection.	100% of tubes in CEA shroud assemblies (Note 2). See Figure 4-18.
MRP-227 Revision 1 entries **Control Element Assembly** C11.1.Remaining instrument guide tubes	All plants with instrument guide tubes in the CEA shroud assembly	Cracking (SCC, Fatigue) that results in missing supports or separation at the welded joint between the tubes and supports.	C11.Peripheral instrument guide tubes within the CEA shroud assemblies	Visual (VT-3) examination. Re-inspection every 10 years following initial inspection.	100% of tubes in CEA shroud assemblies (minimum of 75% of the total population of remaining instrument guide tubes) focusing on the supports and the connecting welds between the supports and the guide tube and CEA shroud. See Figure 4-33.

TABLE 4-5: CE plants Expansion Items

Item	Applicability	Effect (Mechanism)	Primary Link	Exam. Meth./ Freq.	Exam. Coverage
MRP-227 Revision 1-A entries **Control Element Assembly** C11.1.Remaining instrument guide tubes	Entry remained unchanged within MRP-227 Revision 1-A.				
MRP-227 Revision 1-A entries **Lower Support Structure** C8.Core support columns	Plants with full-height bolted or half-height welded core shroud plates	Cracking (SCC, IASCC, Fatigue including damaged or fractured material) Aging Management (IE, TE)	C6.Middle Girth Weld (MGW)	Visual (VT-3) examination Reinspection every 10 years following initial inspection.	Plants with full-height bolted core shroud plates: 25% of the total number of column assemblies (both visible and non-visible from above the core support plate) using a VT-3 examination from above the core support plate. The inspection coverage must be evenly distributed across the population of column assemblies. Plants with core shrouds assembled in two vertical sections: 25% of the accessible surfaces of the core support column welds, from the top side of the core support plate. The inspection coverage must be evenly distributed across the population of core support column welds. (Notes 3 and 4) See Figure 4-36.

TABLE 4-5: CE plants Expansion Items

Existing MRP-227-A entries

Notes to Table 4-5:

1. Examination acceptance criteria and expansion criteria for the CE components are in Table 5-2.
2. A minimum of 75% coverage of the entire examination area or volume, or a minimum sample size of 75% of the total population of like components of the examination is required (including both the accessible and inaccessible portions).

MRP-227 Revision 1 entries

Notes to Table 4-5:

1. Examination acceptance criteria and expansion criteria for the CE components are in Table 5-2.
2. Examination coverage requires examination of either the ID or the OD of the weld.

MRP-227 Revision 1-A entries

Notes to Table 4-5:

1. Examination acceptance criteria and expansion criteria for the CE components are in Table 5-2.
2. Examination coverage requires examination of either the ID or the OD of the weld.
3. The stated minimum coverage requirement is the minimum if no significant indications are found. However, the Examination Acceptance criteria in Section 5 require that additional coverage must be achieved in the same outage if significant flaws are found. This contingency should be considered for inspection planning purposes.
4. Justification that adequate distribution of the inspection coverage has been achieved can be based on geometric or layout arguments. Possible examples include, but are not limited to, inspection of all column assemblies or accessible core support column welds in one quadrant of the core support plate (based on the azimuthal symmetry of the plate) or inspecting every fourth column or weld across the entire plate.
5. Examination coverage requires a minimum of 75% of the weld length for either the ID or the OD of the weld being examined.

TABLE 4-6: Westinghouse plants Expansion Items

Item	Applicability	Effect (Mechanism)	Primary Link	Exam. Meth./ Freq.	Exam. Coverage
Existing MRP-227-A entries					
Upper Internals Assembly Upper core plate	All plants	Cracking (Fatigue, Wear)	CRGT lower flange weld	Enhanced visual (EVT-1) examination. Re-inspection every 10 years following initial inspection.	100% of accessible surfaces (Note 2).
MRP-227 Revision 1 entries					
Upper Internals Assembly W4.1.Upper core plate	All plants	Cracking (Fatigue), Wear, Aging Management (IE)	W4.Lower Girth Weld (LGW)	Visual (VT-3) examination. Re-inspection every 10 years following initial inspection.	Minimum of 25% of core side surfaces. See Figure 4-19.
MRP-227 Revision 1-A entries					
Upper Internals Assembly W4.1.Upper core plate	All plants	Cracking (Fatigue), Wear, Aging Management (IE)	W4.Lower Girth Weld (LGW)	Visual (VT-3) examination. Reinspection every 10 years following initial inspection.	Minimum of 25% of core side surfaces. (Note 3) See Figure 4-19.
Existing MRP-227-A entries					
Lower Internals Assembly Lower support forging or castings	All plants	Cracking Aging Management (TE in Casting)	CRGT lower flange weld	Enhanced visual (EVT-1) examination. Re-inspection every 10 years following initial inspection.	100% of accessible surfaces (Note 2). See Figure 4-33.

TABLE 4-6: Westinghouse plants Expansion Items

Item	Applicability	Effect (Mechanism)	Primary Link	Exam. Meth./ Freq.	Exam. Coverage
MRP-227 Revision 1 entries **Lower Internals Assembly** W3.4.Lower support forging or castings	All plants	Cracking (SCC) Aging Management (TE in Casting)	W3.Upper Core Barrel Flange Weld (UFW)	Visual (VT-3) examination. Re-inspection every 10 years following initial inspection.	25% of bottom (non-core side) surface. See Figure 4-20.
MRP-227 Revision 1-A entries **Lower Internals Assembly** W3.4.Lower support forging or castings	All plants	Cracking (SCC) Aging Management (TE in Casting)	W3.Upper Core Barrel Flange Weld (UFW)	Visual (VT-3) examination. Reinspection every 10 years following initial inspection.	Minimum of 25% of bottom (non-core side) surface. (Note 3) See Figure 4-20.
Existing MRP-227-A entries **Core Barrel Assembly** Barrel-former bolts	All plants	Cracking (IASCC, Fatigue) Aging Management (IE, Void Swelling and ISR)	Baffle-former bolts	Volumetric (UT) examination. Re-inspection every 10 years following initial inspection.	100% of accessible bolts. Accessibility may be limited by presence of thermal shields or neutron pads (Note 2). See Figure 4-23.
MRP-227 Revision 1 entries **Core Barrel Assembly** W6.1.Barrel-former bolts	All plants	Cracking (IASCC, Fatigue) Aging Management (IE, Void Swelling and ISR)	W6.Baffle-former bolts	Volumetric (UT) examination. Re-inspection every 10 years following initial inspection.	100% of accessible barrel-former bolts (Minimum of 75% of the total population). Accessibility may be limited by presence of thermal shield or neutron pads. See Figure 4-21.

TABLE 4-6: Westinghouse plants Expansion Items

Item	Applicability	Effect (Mechanism)	Primary Link	Exam. Meth./ Freq.	Exam. Coverage
MRP-227 Revision 1-A entries **Core Barrel Assembly** W6.1.Barrel-former bolts	Entry remained unchanged within MRP-227 Revision 1-A.				
Existing MRP-227-A entries **Lower Support Assembly** Lower support column bolts	All plants	Cracking (IASCC, Fatigue) Aging Management (IE and ISR)	Baffle-former bolts	Volumetric (UT) examination. Re-inspection every 10 years following initial inspection.	100% of accessible bolts or as supported by plant-specific justification (Note 2). See Figures 4-32 and 4-33.
MRP-227 Revision 1 entries **Lower Support Assembly** W6.2.Lower support column bolts	All plants	Cracking (IASCC, Fatigue) Aging Management (IE and ISR)	W6.Baffle-former bolts	Volumetric (UT) examination. Re-inspection every 10 years following initial inspection.	100% of accessible LSC bolts (Minimum of 75% of the total population) or as supported by plant-specific justification.
MRP-227 Revision 1-A entries **Lower Support Assembly** W6.2.Lower support column bolts	Entry remained unchanged within MRP-227 Revision 1-A.				

TABLE 4-6: Westinghouse plants Expansion Items

Item	Applicability	Effect (Mechanism)	Primary Link	Exam. Meth./ Freq.	Exam. Coverage
Existing MRP-227-A entries **Core Barrel Assembly** Core barrel outlet nozzle welds *(NOTE: This expansion item was deleted and replaced with three (3) expansion items shown below: UGW, UAW, LFW)*	All plants	Cracking (SCC, Fatigue) Aging Management (IE of lower sections)	Upper core barrel flange weld	Enhanced visual (EVT-1) examination. Re-inspection every 10 years following initial inspection.	100% of one side of the accessible surfaces of the selected weld and adjacent base metal (Note 2). See Figure 4-22.
MRP-227 Revision 1 entries **Core Barrel Assembly** W3.1.Upper Girth Weld (UGW)	All plants	Cracking (SCC)	W3.Upper Core Barrel Flange Weld (UFW)	Enhanced visual (EVT-1) examination. Re-inspection every 10 years following initial inspection.	A minimum of 75% of the circumferential surface of the UGW and adjacent base metal shall be examined (Note 2). See Figure 4-13.
MRP-227 Revision 1-A entries **Core Barrel Assembly** W3.1.Upper Girth Weld (UGW)	All plants	Cracking (SCC)	W3.Upper Core Barrel Flange Weld (UFW)	Enhanced visual (EVT-1) examination. Reinspection every 10 years following initial inspection.	100% of the accessible weld length of one side of the UGW and ¾" of adjacent base metal shall be examined (Notes 2 and 5). See Figure 4-13.

TABLE 4-6: Westinghouse plants Expansion Items

Item	Applicability	Effect (Mechanism)	Primary Link	Exam. Meth./ Freq.	Exam. Coverage
MRP-227 Revision 1 entries **Core Barrel Assembly** W3.2.Upper Axial Weld (UAW)	All plants	Cracking (SCC)	W3.Upper Core Barrel Flange Weld (UFW)	Enhanced visual (EVT-1) examination. Re-inspection every 10 years following initial inspection.	A minimum of 75% of the longitudinal surface of the UAW and adjacent base metal shall be examined (Note 2). See Figure 4-13.
MRP-227 Revision 1-A entries **Core Barrel Assembly** W3.2.Upper Axial Weld (UAW)	All plants	Cracking (SCC)	W3.Upper Core Barrel Flange Weld (UFW)	Enhanced visual (EVT-1) examination. Reinspection every 10 years following initial inspection.	100% of the accessible weld length of one side of the UAW and ¾" of adjacent base metal shall be examined (Notes 2 and 5).
MRP-227 Revision 1 entries **Core Barrel Assembly** W3.3.Lower Flange Weld (LFW)	All plants	Cracking (SCC)	W3.Upper Core Barrel Flange Weld (UFW)	Enhanced visual (EVT-1) examination. Re-inspection every 10 years following initial inspection.	A minimum of 75% of the circumferential OD surface of the LFW and adjacent base metal shall be examined (Note 2). See Figure 4-13.

TABLE 4-6: Westinghouse plants Expansion Items

Item	Applicability	Effect (Mechanism)	Primary Link	Exam. Meth./ Freq.	Exam. Coverage
MRP-227 Revision 1-A entries **Core Barrel Assembly** W3.3.Lower Flange Weld (LFW)	All plants	Cracking (SCC)	W3.Upper Core Barrel Flange Weld (UFW)	Enhanced visual (EVT-1) examination. Reinspection every 10 years following initial inspection.	100% of the accessible weld length of the OD surface of the LFW and ¾" of adjacent base metal shall be examined (Note 5). See Figure 4-13.
Existing MRP-227-A entries **Core Barrel Assembly** Upper and lower core barrel cylinder axial welds	All plants	Cracking (SCC, IASCC) Aging Management (IE)	Upper and lower core barrel cylinder girth welds	Enhanced visual (EVT-1) examination. Re-inspection every 10 years following initial inspection.	100% of one side of the accessible surfaces of the selected weld and adjacent base metal (Note 2). See Figure 4-22.
MRP-227 Revision 1 entries **Core Barrel Assembly** W4.2.Middle Axial Welds (MAW) and W4.3.Lower Axial Welds (LAW)	All plants	Cracking (SCC, IASCC) Aging Management (IE)	W4.Lower Girth Weld (LGW)	Enhanced visual (EVT-1) examination. Re-inspection every 10 years following initial inspection	A minimum of 75% of the longitudinal OD surface of all MAW and LAW including the adjacent base metal shall be examined (Note 2). See Figure 4-13.

TABLE 4-6: Westinghouse plants Expansion Items

Item	Applicability	Effect (Mechanism)	Primary Link	Exam. Meth./ Freq.	Exam. Coverage
MRP-227 Revision 1-A entries **Core Barrel Assembly** W4.2.Middle Axial Welds (MAW) and W4.3.Lower Axial Welds (LAW)	All plants	Cracking (SCC, IASCC) Aging Management (IE)	W4.Lower Girth Weld (LGW)	Enhanced visual (EVT-1) examination. Reinspection every 10 years following initial inspection	100% of the accessible weld length of the OD of the MAW and LAW and ¾" of adjacent base metal shall be examined (Notes 5 and 6). See Figure 4-13.
Existing MRP-227-A entries **Lower Support Assembly** Lower support column bodies (non cast)	All plants	Cracking (IASCC) Aging Management (IE)	Upper core barrel flange weld	Enhanced visual (EVT-1) examination. Re-inspection every 10 years following initial inspection.	100% of accessible surfaces (Note 2). See Figure 4-34.
AND **Lower Support Assembly** Lower support column bodies (cast) ***NOTE***: *The items for Lower support column bodies of both cast and non-cast has been combined into a single item as shown below (W4.4).*	All plants	Cracking (IASCC) including the detection of fractured support columns Aging Management (IE)	Control rod guide tube (CRGT) lower flanges	Visual (EVT-1) examination. Re-inspection every 10 years following initial inspection.	100% of accessible support columns (Note 2). See Figure 4-34.
MRP-227 Revision 1 entries **Lower Support Assembly** W4.4.Lower support column bodies (both cast and non-cast)	All plants	Cracking (IASCC), Aging Management (IE)	W4.Lower Girth Weld (LGW)	Visual (VT-3) examination. Re-inspection every 10 years following initial inspection.	25% of column assemblies as visible using a VT-3 examination from above the lower core plate. See Figure 4-23.

TABLE 4-6: Westinghouse plants Expansion Items

Item	Applicability	Effect (Mechanism)	Primary Link	Exam. Meth./ Freq.	Exam. Coverage
MRP-227 Revision 1-A entries **Lower Support Assembly** W4.4.Lower support column bodies (both cast and non-cast)	All plants	Cracking (IASCC), Aging Management (IE)	W4.Lower Girth Weld (LGW)	Visual (VT-3) examination. Reinspection every 10 years following initial inspection.	25% of the total number of column assemblies (both visible and non-visible from above the lower core plate) using a VT-3 examination from above the lower core plate. The inspection coverage must be evenly distributed across the population of column assemblies (Notes 3 and 4). See Figure 4-23.
Existing MRP-227-A entries **Bottom Mounted Instrumentation System** Bottom-mounted instrumentation (BMI) column bodies	All plants	Cracking (Fatigue) including the detection of completely fractured column bodies Aging Management (IE)	Control rod guide tube (CRGT) lower flanges	Visual (VT-3) examination of BMI column bodies as indicated by difficulty of insertion/withdrawal of flux thimbles. Re-inspection every 10 years following initial inspection. Flux thimble insertion/withdrawal to be monitored at each inspection interval.	100% of BMI column bodies for which difficulty is detected during flux thimble insertion/withdrawal. See Figure 4-35.

TABLE 4-6: Westinghouse plants Expansion Items

Item	Applicability	Effect (Mechanism)	Primary Link	Exam. Meth./ Freq.	Exam. Coverage
MRP-227 Revision 1 entries **Bottom Mounted Instrumentation System** W2.2.Bottom-mounted instrumentation (BMI) column bodies	All plants	Cracking (Fatigue) including the detection of completely fractured column bodies Aging Management (IE)	W2.CRGT Lower Flange Welds	Visual (VT-3) examination. Re-inspection every 10 years following initial inspection.	100% of BMI column bodies for which difficulty is detected during flux thimble insertion/withdrawal. See Figure 4-24.
MRP-227 Revision 1-A entries **Bottom Mounted Instrumentation System** W2.2.Bottom-mounted instrumentation (BMI) column bodies	Entry remained unchanged within MRP-227 Revision 1-A.				
MRP-227 Revision 1 entries **Control Rod Guide Tube Assembly** W2.1.Remaining CRGT lower flange welds (NOTE: this expansion item was added for Revision 1 as a clarifying item)	All plants	Cracking (SCC, Fatigue) Aging Management (IE and TE)	W2.CRGT Lower Flange Welds	Enhanced visual (EVT-1) examination to determine the presence of crack-like surface flaws in flange welds. Subsequent examination on a ten-year interval.	A minimum of 75% of the CRGT assembly lower flange weld surfaces and the adjacent base metal for the flange welds not inspected under the primary link.

TABLE 4-6: Westinghouse plants Expansion Items

Item	Applicability	Effect (Mechanism)	Primary Link	Exam. Meth./ Freq.	Exam. Coverage
MRP-227 Revision 1-A entries **Control Rod Guide Tube Assembly** W2.1.Remaining CRGT lower flange welds (NOTE: this expansion item was added for Revision 1 as a clarifying item)	All plants	Cracking (SCC, Fatigue) Aging Management (IE and TE)	W2.CRGT Lower Flange Welds	Enhanced visual (EVT-1) examination to determine the presence of crack-like surface flaws in flange welds. Subsequent examination on a ten-year interval.	A minimum of 75% of the CRGT assembly lower flange weld surfaces and 0.25-inch of the adjacent base metal for the flange welds not inspected under the primary link.

Existing MRP-227-A entries

Notes to Table 4-6:

1. Examination acceptance criteria and expansion criteria for the Westinghouse components are in Table 5-3.
2. A minimum of 75% coverage of the entire examination area or volume, or a minimum sample size of 75% of the total population of like components of the examination is required (including both the accessible and inaccessible portions).

MRP-227 Revision 1 entries

Notes to Table 4-6:

1. Examination acceptance criteria and expansion criteria for the Westinghouse components are in Table 5-3.
2. Examination coverage requires examination of either the ID or the OD of the weld.

MRP-227 Revision 1-A entries

Notes to Table 4-6:

1. Examination acceptance criteria and expansion criteria for the Westinghouse components are in Table 5-3.
2. Examination coverage requires examination of either the ID or the OD of the weld.
3. The stated minimum coverage requirement is the minimum if no significant indications are found. However, the Examination Acceptance criteria in Section 5 require that additional coverage must be achieved in the same outage if significant flaws are found. This contingency should be considered for inspection planning purposes.
4. Justification that adequate distribution of the inspection coverage has been achieved can be based on geometric or layout arguments. Possible examples include, but are not limited to, inspection of all column assemblies in one quadrant of the lower core plate (based on the azimuthal symmetry of the plate) or inspecting every fourth column across the entire plate.
5. A minimum coverage of 75% of the weld length on the surface being examined shall be achieved; however, for welds with limited access (Note 6), a minimum examination coverage of 50% of the weld length on the surface being examined shall be achieved.
6. Accessibility to the MAW and LAW may be limited by the thermal shield or neutron panels – no disassembly to achieve higher weld length coverage is required.

Appendix D: Requests for Additional Information and MRP Responses

EPRI/Industry Responses Letters to US NRC Requests for Additional Information (RAI) on MRP-227-Revision 1:

MRP 2017-027, dated 10/16/2017
MRP 2018-003, dated 1/30/2018
MRP 2018-011, dated 5/17/2018
MRP 2018-026, dated 9/28/2018

DATE: October 16, 2017

TO: Document Control Desk, Attn: Joe Holonich
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555-001

FROM: Mike Hoehn II, Ameren Missouri, MRP Integration Committee Chairman
Brian Burgos, EPRI, MRP Program Manager

SUBJECT: RESPONSES TO NRC REQUEST FOR ADDITIONAL INFORMATION FOR ELECTRIC POWER RESEARCH INSTITUTE TOPICAL REPORT MRP-227, REVISION 1, "MATERIALS RELIABILITY PROGRAM: PRESSURIZED WATER REACTOR INTERNALS INSPECTION AND EVALUATIONS GUIDELINE"
(CAC NO. MF7740)

Dear Sir:

This letter provides industry responses to the NRC staff's requests for additional information (RAI) related to MRP-227 Revision 1, entitled "Materials Reliability Program: Pressurized Water Reactor Internals Inspection and Evaluations Guideline." EPRI received the formal RAIs from the NRC in a letter dated 5/15/2017 [ADAMS accession number ML17079A027]. EPRI discussed preliminary responses to the RAIs with NRC staff during public meetings on 7/12-13/2017 [ML17159A432], 9/6/2017 [ML17248A542], and 10/5/2017 [ML17278A034].

The attachments to this letter provide the industry's responses to a portion of the RAIs (Table 1), and work continues on formalizing responses to the remaining RAIs by the end of 2017. Enclosed are four (4) copies of this letter and the attachments.

If you have additional questions or require further information, please contact Kyle Amberge (kamberge@epri.com, (704) 595-2039) or Brian Burgos (bburgos@epri.com, (724) 610-8559) or myself.

Sincerely,

Mike Hoehn II, Ameren-Missouri MRP Integration Committee Chair	Brian Burgos, Program Manager Materials Reliability Program

Project No. 669
cc: Jeff Poehler, NRC-NRR

Table 1

EPRI Proposed Clarifications and Editorial Changes to MRP-227, Rev.1 Based on RAI Responses

RAI #	RAI Subject	MRP-227, R1 Table Updated	Reason for Change
6	Core Barrel Bolting Locking Devices	Table 4-1	Clarification of applicable aging degradation mechanisms for locking devices and locking device welds.
7	WEC CB Lower Girth weld expansion coverage if indication found	Table 5-3	It should have been noted that a surface breaking indication found would require expansion to 100% of the accessible locations of the lower girth weld, consistent with other welds.
8	Baffle Bolt Operating Experience	Table 4-3	Table 4-3 will be updated to incorporated and/or reference the BFB Interim Guidance.
14	WEC Upper Core Plate and Lower Support Forging/Casting	Table 4-6 Table 5-3	Clarification that an indication found in the initial 25% coverage exam would require expansion to 100% of the accessible surface of the side of plate to be examined.
15	Split Pin Classification and Aging Management	Section 4.4 Section 4.5	Discussion on split pins is moved from Section 4.4 to 4.5 to avoid confusion on the classification of the component – which is "No Additional Measures." Also, re-stated in similar language that X-750 split pins should be managed (or replaced) by the plant owners and clarified that split pin degradation is an asset management issue and not a safety concern.
19	Guide Card Aging Management	Table 4-3	Remove revision number from WCAP-17451 reference in Table 4-3 to reference the most current approved version.
20	Examination of Adjacent Base Metal	Table 4-6	The specified base metal examination coverage was unintentionally omitted and should be the same as the matching component in Table 4-3
23	Inspection Technique for the CE Core Shroud Assembly	Table 5-2	VT-1 should have been identified as the inspection technique, consistent with Table 4-2.

ATTACHMENT 1 – Industry Responses to RAIs Related to WEC/CE-Design PWRs

(Ref. Westinghouse letter LTR-RIAM-17-37)

ATTACHMENT 1

Responses to NRC Requests for Additional Information (RAIs) on MRP-227, Revision 1

1.0 Background and Introduction

This report provides responses to the U.S. Nuclear Regulatory Commission (NRC) Requests for Additional Information (RAIs) [1] related to the staff's review [2] of MRP-227, Revision 1 [3].

2.0 References

1. U.S. Nuclear Regulatory Commission Letter, "Request for Additional Information for Electric Power Research Institute Topical Report MRP-227, Revision 1, 'Materials Reliability Program: Pressurized Water Reactor Internals Inspection and Evaluation Guideline' (CAC No. MF7740)" May 15, 2017 (ADAMS Accession No. ML17079A027).
2. U.S. Nuclear Regulatory Commission Letter, "Acceptance Review of Electric Power Research Institute Topical Report MRP-227, Revision 1, 'Materials Reliability Program: Pressurized Water Reactor Internals Inspection and Evaluations Guideline,' (TAC NO. MF7740)," July 15, 2016 (ADAMS Accession No. ML16154A063).
3. *Materials Reliability Program: Pressurized Water Reactor Internals Inspection and Evaluation Guidelines (MRP-227, Revision 1)*. EPRI, Palo, Alto, CA: 2015. 3002005349.
4. Westinghouse Nuclear Safety Advisory Letter, NSAL-16-1, Revision 1, "Baffle-Former Bolts," Westinghouse Electric Company LLC, August 1, 2016.
5. Letter from Bernie Rudell and Anne Demma to the NRC, Subject: "Transmittal of NEI-03-08, 'Needed' Interim Guidance Regarding Baffle Former Bolt inspections for Tier 1 plants as Defined in Westinghouse NSAL-16-01," EPRI Materials Reliability Program, transmitted by letter MRP 2016-022, July 27, 2016 (ADAMS Accession No. ML16211A054).
6. Letter from Bernie Rudell and Brian Burgos to the NRC, Subject: Transmittal of NEI-03-08 "Needed" Interim Guidance Regarding Baffle Former Bolt inspections for U.S. PWR plants as Defined in Westinghouse NSAL 16-01, MRP Letter 2017-011, March 23, 2017 (ADAMS Accession No. ML17087A107).
7. WCAP-17096-NP-A, Rev 2, "Reactor Internals Acceptance Criteria Methodology and Data Requirements." August 31, 2016 (ADAMS Accession No. ML16279A320)

8. Westinghouse Nuclear Safety Advisory Letter, NSAL-17-1, Revision 0, "Guide Tube Guide Card Wear Attributed to Ion Nitride Rod Cluster Control Assembly," Westinghouse Electric Co. LLC, January 16, 2017.

9. WCAP-17451-P, Rev. 1, "Reactor Internals Guide Tube Wear - Westinghouse Domestic Fleet Operational Projects," October 31, 2013 (ADAMS Accession No. ML15041A106)

10. Letter from David Czufin and Brian Burgos to the MRP Integration Committee Members, Subject: Transmittal of NEI-03-08 "Needed" Interim Guidance Regarding Baffle Former Bolt Inspections for PWR Plants as Defined in Westinghouse NSAL 16-01 Rev. 1, MRP Letter 2017-009, March 15, 2017 (ADAMS Accession No. ML17087A106).

11. Letter from Bernie Rudell and Anna Demma to the MRP Members, Subject: Transmittal of NEI 03-08 "Needed" Interim Guidance Regarding Baffle Former Bolt Inspections for Tier 1 plants as Defined in Westinghouse NSAL 16-01, MRP Letter 2016-021, July 25, 2016 (ADAMS Accession No. ML16211A054).

12. Pressurized Water Reactor Owners Group Report, PWROG-15032-NP, "PA-MSC-1288 Statistical Assessment of PWR RV Internals CASS Materials," November 2015.

13. Nuclear Regulatory Commission Safety Evaluation, "Final Safety Evaluation of the BWRVIP-234: Thermal Aging and Neutron Embrittlement Evaluation of Cast Austenitic Stainless Steel for BWR Internals (TAC No. ME5060), June 22, 2016, ML16096A002.

14. Nuclear Regulatory Commission letter from Christopher I. Grimes : 'License Renewal Issue No 98-003, 'Thermal Aging Embrittlement of Cast Austenitic Stainless Steel Components"' May 19, 2000, ML003717179

15. Nuclear Regulatory Commission Letter, "List of Questions Related to Materials Reliability Program 2017-009 , Attachment to Materials Reliability Letter 2017-011, 'Transmittal of "Needed" Interim Guidance Regarding Baffle Former Bolt Inspections for U.S. Pressurized-Water Reactor Plants as Defined in Westinghouse Nuclear Safety Advisory Letter 16-01,' Dated March 23, 2017," April 3, 2017 (ADAMS Accession No. ML17095A886)

16. Letter from M. Hoehn II and B. Burgos to NRC Document Control Desk, Subject: Transmittal of Engineering Tools Supporting Utility Planning for Baffle Former Bolt Inspections," MRP Letter 2017-019, August 4, 2017 (ADAMS Accession No. ML17222A166).

17. Letter from Mike Hoehn II and Brian Burgos to NRC Document Control Desk, Subject: Responses to the Questions from the U.S. Nuclear Regulatory Commission Staff on the Baffle-Former Bolt 'Needed' Guidance Transmitted in Letter MRP 2017-009, MRP Letter 2017-015, July 13, 2017 (ADAMS Accession No. ML17261B149).

18. Letter from B. Rudell and B. Burgos to the MRP Integration Committee, Subject: Summary 'White Paper' of the Baffle-Former Bolt Prediction Results Provided by Structural Integrity Associates, AREVA and Westinghouse, MRP Letter 2017-010, March 17, 2017 (ADAMS Accession No. ML17222A169)

19. Letter from Mike Hoehn II and Brian Burgos to the MRP Integration Committee Members, Subject: Transmit Baffle-Former-Bolt Response Options 'Playbook' for Information and Use in Planning for PWR Refueling Outages, MRP Letter 2017-013, May 12, 2017 (ADAMS Accession No. ML17222A167)

20. Westinghouse Report, WCAP-14577-A, Rev. 1, "License Renewal Evaluation: Aging Management for Reactor Internals," March 2001 (ADAMS Accession No. ML003779006).

3.0 RAI Responses

Responses to individual NRC RAIs are provided in this section. Each section contains the RAI exactly as transmitted by the NRC, followed by the proposed response.

This letter transmits responses to the NRC request for additional information (RAI) (issued via [11]) related to MRP-227, Revision 1 [3]. Responses to the following RAIs are included in this transmittal: 7, 8, 13, 14, 15, 19, 20, 23, 24, 26 and 27.

Responses to RAIs 1-4, 6, 11, 17-18, 21-22, and 25 are not provided, as they are specific to B&W plants.

Responses to RAIs 5, 9, 10, 12 and 16 will be provided in a separate transmittal because these responses include recommended markups to MRP-227, Revision 1 as a means to address the RAI. These markups were considered changes to the MRP-227, Revision 1 guidance, as opposed to just clarifications, and as such require further review by the EPRI Executive Oversight Committee (PMMP). The additional review is required because the recommended changes affect the Section 4 and 5 tables of MRP-227, Revision 1, which are identified as an NEI-03-08 "Needed" requirement.

3.1 Response to NRC RAI 7

3.1.1 NRC Question

For all the welds listed in Table 1 except for Item W4., "Core Barrel Assembly – Lower Girth Weld (LGW)," the examination acceptance and expansion criteria in Table 5-2 and Table 5-3 require the inspection coverage to be extended to include 100% of the accessible length of the weld during the same refueling outage, if there is confirmed detection of a surface breaking linear indication in that weld. Should this expansion also be applied to Item W4? If not, provide a technical justification.

3.1.2 Industry Response

The examination acceptance and expansion criteria for Item W4, Core Barrel Assembly – Lower Girth Weld will be modified to require the inspection coverage to be extended to include 100% of the accessible length of the weld during the same refueling outage, if there is confirmed detection of a surface breaking linear indication in that weld.

The changes to be made to Table 5-3 in MRP-227, Revision 1 are shown below in red.

Primary Item	Applicability	Examination Acceptance Criteria	Expansion Link(s)	Expansion Criteria	Expansion Item Examination Acceptance Criteria
Core Barrel Assembly Lower girth weld (LGW)	All plants	Periodic enhanced visual (EVT-1) examination. The specific relevant condition is a detectable crack-like surface indication.	Upper core plate Lower support column bodies (cast and non-cast) Middle axial welds (MAW) Lower axial welds (LAW)	a. The confirmed detection of a surface-breaking linear indication in the LGW shall require that inspection coverage of the LGW be extended to include 100% of the accessible length during the same outage. ~~a.~~ b. The confirmed detection and sizing of a surface-breaking indication with a length greater than two inches in the LGW shall require inspection of the upper core plate within the next three refueling outages. ~~b.~~ c. The confirmed detection and sizing of a surface-breaking indication with a length greater than two inches in the LGW shall require inspection of the lower support column bodies (cast and non-cast) within the next three refueling cycles. ~~c.~~ d. The confirmed detection and sizing of a surface-breaking indication with a length greater than two inches in the LGW shall require that the inspections be expanded to include the lower core barrel cylinder axial welds by the completion of the next refueling outage.	a. The specific relevant condition for the remaining LGW length is a detectable crack-like surface indication. ~~a.~~ b. The specific relevant conditions for the inspection of the upper core plate are broken or missing parts of the plate. ~~b.~~ c. The specific relevant conditions for the inspection of the lower support column bodies (cast or non-cast) are fractured, misaligned, or missing columns. ~~c.~~ d. The specific relevant condition for the expansion MAW and LAW inspections is a detectable crack-like surface indication.

3.2 Response to NRC RAI 8

3.2.1 NRC Question

Operating experience (OE) in 2016 showed that Westinghouse 4-loop design plants operating in a downflow configuration with Type 347 stainless steel baffle-former bolts (BFB) experienced higher-than–expected levels of degradation of BFB, and also significant clustering of degraded bolts. However, MRP-227, Rev. 1 does not include any changes in the guidance for BFB from MRP-227-A.

Westinghouse Nuclear Safety Advisory Letter NSAL-16-1, Revision 1 "Baffle-Former Bolts (Ref. 2)," categorized all Westinghouse and CE design RVI with respect to susceptibility to BFB degradation. EPRI interim guidance in MRP Letter 2016-022 (Ref. 3) endorsed the recommendation of Westinghouse NSAL 16-1 that 4-loop, downflow plants with Type 347 bolts complete baseline UT examinations of BFB by the next refueling outage. These baseline examinations are expected to be complete by the end of 2017. EPRI interim guidance in MRP Letter 2017-002 (Ref. 4) endorsed the NSAL 16-1 guidance for 2-loop and 3-loop downflow plants (Tier 2), which calls for a review of previous UT examination results from these plants for evidence of clustering, with UT examination the next refueling outage if evidence of clustering is seen. The EPRI interim guidance does not provide any guidance on how the subsequent examination interval is to be determined for Tier 1 and Tier 2 plants. The default subsequent examination interval in MRP-227, Revision 1, remains ten years. However, the NRC staff is concerned that a default subsequent examination interval of ten years may not be appropriate for the highest susceptibility groups of plants.

If BFB degradation is found, an engineering evaluation is required. MRP-227, Revision 1, Section 7.5 defines as NEI 03-08 "needed" guidance that, if 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, and that such engineering evaluations shall be conducted in accordance with NRC-approved evaluation methods (i.e. ASME Code Section XI, WCAP-17096-NP or equivalent method). Current NRC-approved guidance for determining the subsequent examination interval for BFBs is found in WCAP-17096-NP-A, Rev. 2 "Reactor Internals Acceptance Criteria Methodology and Data Requirements" (Ref. 5), pages E-42 to E-43, which allows a subsequent examination interval of 10 years provided that no more of 50 percent of the initial margin with respect to the minimum required number of bolts is found degraded at the initial UT examination. However, WCAP-17096-NP-A does not provide any guidance for determining the subsequent examination interval if greater than 50 percent of the bolts constituting the margin are degraded, even if degraded bolts are replaced. In addition, the guidance in WCAP-17096-NP-A for determining the subsequent examination interval does not take into account the possibility of clustering of degraded bolts as was seen in the 4-loop plants in 2016, and did not account for the large extent of BFB degradation seen in certain plants.

The NRC staff requests that EPRI:

a. Discuss whether revised guidance for BFB needs to be incorporated into MRP-227, Rev. 1. If not, why not?

b. If such guidance should be incorporated, provide specifics on the initial examination coverage and schedule, and on how the subsequent examination coverage and timing would be determined.

c. Considering the recent OE with BFB degradation, justify that a ten-year subsequent examination interval remains appropriate for BFB. This justification should consider the possible effects of clustering.

d. How will the schedule for subsequent examination be determined if examination results show that greater than 50 percent of the numerical margin of bolts is degraded?

e. Provide a justification that the criteria allowing subsequent examination of BFB may be performed in ten years, provided 50 percent or less of the numerical margin of BFB is degraded, is still appropriate considering the discovery of clustering of degraded BFB, and the discovery of more extensive BFB degradation than expected.

3.2.2 Industry Response

a. The NEI-03-08 "Needed" Interim Guidance regarding baffle-former bolt inspections developed by the industry and summarized in EPRI MRP letters MRP-2016-021 [11] (transmitted to the NRC via MRP 2016-022, ML16211A054) [5] and MRP 2017-009 [10] (transmitted to the NRC via MRP 2017-011, ML17087A107) [6] will be incorporated into MRP-227, Revision 1, including responses to the staff's questions on the interim guidance [15] (responses issued in [17]). The BFB interim guidance letter [10] references NSAL-16-1, Revision 1 (transmitted to the NRC via ML16222A513) [4] for the updated inspection requirements which follow a tiered system based on the plant's configuration.

   As discussed in MRP 2017-015 [17], the guidance relating to the re-inspection criteria, specifically MRP 2017-009 [10] Part B (Note b), will be revised by incorporating the Interim Guidance of MRP 2017-009 into the final staff-reviewed version of MRP-227, Revision 1 or by revised guidance for W-ID: 7 in WCAP-17096-NP-A (potentially via interim guidance). The revised guidance would require submittal consistent with guidance imposed for CE-ID: 6 and 7 and W-ID: 3, 3.1, 4 and 5 which states "any proposal for extension of the verification period beyond a single refueling cycle or use of an alternative verification process would require a technical basis to be submitted to the regulator."

   Similar language consistent with the above will be added to the final staff-reviewed version of MRP-227, Revision 1 such that in order to apply the plant-specific evaluation in the context of Note (b) to justify a longer re-inspection interval the evaluation will be submitted to the NRC for information within one (1) year after any BFB inspection or bolt replacement activity for which the results trigger the reduced re-inspection interval. If the evaluation is completed after this one year timeframe it shall be submitted within 90 days of completion of the evaluation.

   It is also noted that the technical approach for management of BFB aging degradation was previously reviewed and endorsed by the NRC via safety evaluation onWCAP-14577-A, Rev. 1 [20].

b. The initial examination coverage and schedule for the baffle-former bolts in MRP-227, Revision 1 will be per the NEI-03-08 "Needed" Interim Guidance published in letters MRP 2016-021 [11] and MRP 2017-009 [10], which reference NSAL-16-1, Revision 1. The inspection schedules for

initial examination in the interim guidance are dependent on the plant tiers assigned by the NSAL.

MRP 2016-021 states:

"All plants identified as Tier 1a plants in Westinghouse NSAL 16-1 shall perform an ultrasonic examination (UT) volumetric inspection of the full population of baffle-former bolts at the next scheduled refueling outage.

Plants identified as Tier 1b in Westinghouse NSAL 16-1 shall complete a visual VT-3 examination of the full population of baffle-former bolts at the next scheduled refueling outage. If degradation is detected the plant shall complete actions consistent with Tier 1a plants, if no degradation is detected during the visual VT-3 examination, an ultrasonic examination (UT) consistent with Tier 1a plants' guidance shall be completed during the second refueling outage after issuance of this interim guidance."

Per MRP 2017-009 [10] and MRP 2017-010[18]:

- Tier 1 plants are to perform the baseline UT examination by approximately 20 EFPY, or the next refueling outage after March 1, 2016.

The evaluation performed in [18] concluded that the baseline inspection for Tier 1 plants should be performed by 20 EFPY. However, plants identified as Tier 1 performed their baseline UT exam by their next refueling outage in accordance with the "Needed" requirement issued by [11].

Per MRP 2017-009:

- Tier 2 plants are to perform the examination at no later than 30 EFPY (initial baseline UT exams performed prior to 1/1/2008 are acceptable).
- All remaining plants are to perform the examination at no later than 35 EFPY.

Re-examination periods shall be determined by plant-specific evaluation per the MRP-227 Needed Requirement 7.5 as documented and dispositioned in the owner's plant corrective action program. If atypical or aggressive baffle-former bolt degradation is observed, the interim guidance provides limitations to the permitted re-inspection interval unless further evaluation is performed to justify a longer interval. The re-inspection interval is not permitted to exceed 10 years.

These requirements are defined in EPRI letters MRP 2016-021 and MRP 2017-009 which will be incorporated into MRP-227, Revision 1 via incorporating the guidance from these interim guidance letters for initial and re-inspection requirements as shown following response E to this RAI.

c. As noted in the response to Part B, limitations have been placed on the 10-year re-examination period if atypical or aggressive baffle-former bolt degradation has been observed. 10 years is an upper limit for re-inspection interval and the exact re-inspection interval must be justified by a plant-specific evaluation if it is exceeds the following defined limits. These limitations have been defined in the NEI-03-08 "Needed" Interim Guidance published by the industry in EPRI MRP letter MRP 2017-009. For downflow plants with indications in 3% or greater of the bolt population, or that demonstrate clustering of indications, the re-inspection period is not to exceed 6 years. For upflow plants with indications in 5% or greater of the bolt population, or that demonstrate clustering, the re-inspection period is not to exceed 6 years. Clustering is defined in MRP 2017-009 as three or more adjacent defective baffle-former bolts or more than 40% defective baffle-former bolts on the same baffle plate. This re-examination period can be extended to 10 years through a plant-specific evaluation that justifies such an extension. These defined limits on re-inspection periods, in addition to being referenced in an updated MRP-227, Revision 1, will be incorporated into a revised version of WCAP-17096-NP-A, Revision 2 [7] (possibly through Interim Guidance).

d. Determination of the subsequent examination interval and application of the 50 percent numerical margin from WCAP-17096-NP-A, Revision 2 [7] are part of developing acceptance criteria for the baffle-former bolts. MRP-227, Revision 1 specifies in Table 5-3 that "The examination acceptance criteria for the UT of the baffle-former bolts shall be established as part of the examination technical justification." In Sections 6 and 7.5 of MRP-227, Revision 1, WCAP-17096-NP-A, Revision 2 is referenced as an acceptable option for performing the acceptance criteria evaluations.

   WCAP-17096-NP-A, Revision 2 is still the applicable document for addressing baffle-former bolt margin requirements, and MRP-227, Revision 1 will still reference that methodology report. Currently, the baffle-former bolt acceptance criteria methodology requires that less than 50% of the initial margin be consumed for operation through the next 10-year inspection interval. A PWROG program is underway to make various updates to WCAP-17096, including potential updates to the baffle-former bolt acceptance criteria methodology. Changes will be made to make the acceptance criteria methodology document consistent with the baffle-former bolt interim guidance issued via EPRI letters MRP 2016-021 and MRP 2017-009. Specifically, the updates will include the re-inspection criteria (based on % of UT indications and the presence of clustering, as described in the response to part C) and associated re-examination intervals.

e. As discussed in responses to parts B and C, limitations have been placed on the 10-year re-examination period by the NEI-03-08 "Needed" Interim Guidance published by the industry in EPRI MRP letter MRP 2017-009 if atypical or aggressive baffle-former bolt degradation has been observed. This places limitations on the re-examination period based on the plant configuration and the level and pattern of degradation observed during the inspection, unless justification by plant-specific evaluation is performed.

   As noted in the response to part D, a PWROG program is currently underway to update WCAP-17096, including potential updates to the baffle-former bolt acceptance criteria methodology based on the contents of letter MRP 2017-009 and the recent operating experience. This program will evaluate the 50 percent margin criterion. This assessment is expected to be completed within the staff's timeframe for development of the final safety evaluation on MRP-227, Revision 1. Once the updates to the document are complete, it will be evaluated in accordance with the NEI-

03-08 Generic Topical Report screening process and, if determined to be appropriate, will be submitted to the NRC for review and safety evaluation.

The following changes will be incorporated into the final NRC-reviewed version of MRP-227, Revision 1:

Primary Item	Applicability	Effect (Mechanism)	Expansion Link (Note 1)	Examination Method/Frequency (Note 1)	Examination Coverage
W6.Baffle-Former Assembly Baffle-former bolts （Note 9）	All plants	Cracking （IASCC, Fatigue） Aging Management （IE and ISR） （Note 6）	Lower support column bolts, Barrel-former bolts	Baseline volumetric （UT） examination ~~between 25 and 35 EFPY,~~ interval is dependent on the plant design （Note 11）. ~~with subsequent~~ Subsequent examination ~~on a ten-year intervalis~~ is dependent on the plant design and the results of the baseline inspection （Note 12）.	100% of accessible bolts. （Note 3） See Figure 4-15.

11. In accordance with MRP 2017-009 and MRP 2017-010, Tier 1 plants are to perform the baseline UT examination by 20 EFPY or during the next refueling outage after March 1, 2016. Per MRP 2017-009, Tier 2 plants are to perform the baseline UT examination at no later than 30 EFPY (initial Tier 2 plant baseline UT exams performed prior to 1/1/2018 are acceptable). All other remaining plants are to perform the baseline UT examination at no later than 35 EFPY.

12. Re-examination periods shall be determined by plant-specific evaluation per the MRP-227 Needed Requirement 7.5 as documented and dispositioned in the owner's plant corrective action program. If atypical or aggressive baffle-former bolt degradation is observed, the interim guidance (MRP 2016-021 and MRP 2017-009) provides limitations to the permitted re-inspection interval (not to exceed 6 years maximum) unless further evaluation is performed to justify a longer interval. If evaluation justifies a longer re-inspection interval, it is not permitted to exceed 10 years.

3.3 Response to NRC RAI 13

3.3.1 NRC Question

The following four areas pertaining to Tables 4-1 through 4-3 could be informed by OE related to the actual accessibility achieved for primary system components during baseline inspections:

1. In MRP-227, Rev. 1, has OE been used to modify or clarify examination coverage requirements of MRP-227-A based on the actual accessibility achieved during the examinations completed to date? If so, identify the components that have had examination coverage revised based on OE, and describe the reason for the change. If coverage requirements have not been revised based on OE, justify why this has not been done.
2. Has OE with actual coverage achieved resulted in any primary component that was previously considered to be accessible being reclassified as inaccessible, either because of the percentage of the component surface area, length, or population that is accessible was insufficient to provide reasonable assurance of functionality, or because insufficient coverage was achieved of the most likely portion of the component to exhibit degradation? Identify any primary components that have been reclassified as inaccessible and identify what alternate measures, such as an engineering analysis, were taken to provide reasonable assurance of component functionality.
3. For primary components reclassified as inaccessible, were the expansion links reevaluated for these components?
4. For any primary components reclassified as inaccessible, were alternate primary components selected?

3.3.2 Industry Response

1. OE has been applied to modifying and re-classifying the inspection requirements for the core barrel welds. During preparation for inspections, more details on the typical naming used for the core barrel welds and more precise locations for the welds were found. This information on naming and location provided the basis for the renamed core barrel weld components listed in MRP-227, Revision 1 and the specific aging-related degradation mechanisms assigned to each weld. This was also an input to the assignment of welds to be primary or expansion components.

   The accessibility of the core barrel girth welds behind a thermal shield or neutron pads was a key reason behind the need to justify a lower amount of coverage on the core barrel weld primary items. The neutron panel is bolted on the core barrel and inspection cannot be completed of sections of the girth welds that are covered by the neutron panel. The exact percentage of the core barrel circumference covered by the neutron panels varies with plant design. Between 50 and 60 percent of the barrel circumference will be accessible to a visual inspection in a neutron panel plant.

Other than the core barrel welds, to date OE has not suggested other changes are needed to modify or clarify examination coverage requirements for other MRP-227-A components for Westinghouse plants based on the actual accessibility achieved during the examinations that have been completed.

The CE RVI-design Expansion component - ribs and rings - has been determined to be inaccessible for inspection. This was based on a further technical review of drawings, and not based on any OE from prior inspections. The inspection technique for the Westinghouse RVI-design upper core plate and lower support forging/casting was changed from EVT-1 to VT-3 and coverage was reduced to 25%. While there are accessibility concerns with these components, the inspection requirements were not changed as a result of OE.

2. No Primary components have been reclassified as inaccessible based on OE.
3. No Primary components have been reclassified as inaccessible based on OE.
4. No Primary components have been reclassified as inaccessible based on OE.

3.4 Response to NRC RAI 14

3.4.1 NRC Question

The inspection method and coverage for two Westinghouse expansion components, the Upper Internals Assembly – Item W4.1., “Upper Core Plate and the Lower Internals Assembly,” – Item W3.4., “lower support forging or casting,” has been changed from EVT-1 examination of 100 percent of accessible surfaces to VT-3 examination of 25% of the bottom (non-core side) surfaces. However, both of these items are non-redundant components. The NRC staff does not generally consider VT-3 examination to be an adequate examination method for non-redundant components unless these components are highly flaw tolerant. In addition, the examination coverage has been reduced. The NRC staff is concerned that the reduced examination coverage is not sufficient to provide reasonable assurance of component functionality, considering that these are high consequence of failure components.

Therefore, the NRC staff requests that EPRI:

a. Justify the use of VT-3 examination for these components;
b. Justify the reduction in examination coverage from 100 percent to 25 percent. The technical justification for the reduction in examination coverage should provide reasonable assurance that (1) the functionality of the components will be maintained and (2) the structural integrity of the components will be maintained to ensure safe shutdown of the reactor during the PEO.
c. Is it intended that if the examination of the 25 percent sample of these items reveals indications, the examination coverage will be expanded to include the remaining accessible surfaces of these components? If not, why not?

3.4.2 Industry Response

a. The examination technique for the upper core plate and lower support forging/casting has been changed to a VT-3 inspection based on consideration of the geometry and functionality of the components. In both cases, the geometry of these components is a large, thick plate with many holes passing through to allow reactor coolant flow. The holes are distributed across the entire plate in both cases. These holes are expected to act as natural crack arresting locations, and postulated cracks are not expected to propagate without encountering one of these holes. Functionality loss would require full section cracking of multiple ligaments (on the upper core plate, for example, there are four flow holes associated with each fuel assembly, including multiple ligaments supporting the hold-down force on that fuel assembly), bent or misaligned ligaments, or missing pieces of the plate. There is a low likelihood of either plate experiencing crack propagation to this extent given the geometry of these components. In addition, an EVT-1 inspection technique is not necessary to identify potential indications that could lead to failure of the component. VT-3 is an adequate technique for a general condition inspection that would identify cracks in the structure that could affect the functionality of the component.

As part of the Revision 1 work there was a deliberate effort to revisit access difficulties with several Expansion components to determine whether the EVT-1 inspection requirement were indeed warranted or whether the visual VT-3 standard, better corresponding to current inspection equipment capabilities, could be justified. Note that there is limited accessibility to perform the inspections on either component, particularly for the lower support forging/casting. The limited access mainly restricts the ability to achieve the EVT-1 camera angle requirement and the proximity required to maintain the maximum distance of the lens to the surface of the item to be examined and minimum zoom that may be used during the examination as required by the resolution demonstration check.

Multiple components are attached to the upper core plate, such as the guide tubes on the top surface and inserts at the alignment pin notches, which obstruct the ability to perform the EVT-1 required under MRP-227-A. The bottom surface of the upper core plate is obstructed when the upper internals are in the storage stand, thus the only way to improve access would be to suspend the upper internals which is highly undesirable. The lower support forging/casting has multiple attached components on the top and bottom surfaces, which would impede the ability to achieve the distance and camera angle required to perform EVT-1 inspections. The top surface of the lower support forging/casting is essentially inaccessible, due to the presence of multiple plates and columns above the lower support and the lack of a viable access path from underneath. Inspection of the bottom surface of the lower support forging/casting is further complicated by its location at the very bottom of the lower internals. These accessibility issues are further reasons for a reduction in the inspection requirements, beyond the technical considerations above.

For the lower support forging/casting, the reduced inspection requirement is also supported by a consideration of the most likely locations for degradation on the component. The lower support forging/casting is attached to the bottom of the core barrel by the lower flange weld (LFW). This weld is expected to be the most likely location associated with the forging/casting to experience cracking first. The weld would be subject to higher stresses due to its geometry (it has a smaller cross-section than the plate) and the presence of the weld (weld residual stresses). An EVT-1 inspection of the LFW is already included in the MRP-227, Revision 1 inspection requirements as an expansion from the core barrel upper flange weld (UFW). Thus, the most susceptible location on the forging/casting will be inspected in the most detail if this expansion occurs.

The reduced inspection requirement for the lower support forging/casting is further supported by a consideration of the likelihood of degradation in the component. In the original MRP-191 degradation mechanism screening, the lower support casting was only screened in for thermal embrittlement (TE) and the lower support forging was not screened in for any degradation mechanisms at all. Both of these components are far enough from the core to not be affected by radiation effects, such as irradiation embrittlement. Thermal embrittlement in the casting has been addressed by the industry through the statistical assessment of PWR Reactor Vessel Internals CASS materials documented in PWROG-15032 [12], which concluded that statistically, the majority of the CASS components in service should have ferrite content below the threshold

for TE. PWROG-15032 further concluded that even fully-aged CASS material with low molybdenum content will not have its toughness reduced below the minimum levels required for acceptable toughness for internals components, whether the 255 kJ/m^2 threshold originally identified in the "Grimes Letter" [14] or the revised level of 200 kJ/m^2 more recently identified in the safety evaluation of BWRVIP-234 [13]. With TE addressed for the casting and no other degradation mechanisms applicable, it is reasonable to perform a general condition inspection to monitor for changes in this component.

VT-3 is an appropriate inspection technique for a general condition inspection looking for the indications mentioned in the functionality discussion above. These conditions could range from multiple completely fractured ligaments between the holes in the plate, displaced or bent locations on the plate, fractured or damaged fuel alignment pins (for the upper core plate), or missing pieces of the plate.

b. A 25% inspection coverage requirement is justified due to the likelihood and functionality bases for these components as discussed in Part A. There is a low likelihood that the type of degradation required to compromise functionality in these components will occur, due to the geometry of the plates and, for the lower support forging/casting, the lack of a driving mechanism for degradation. Any postulated crack is expected to be arrested in one of the many holes in either the upper core plate or the lower support forging/casting. Without gross propagation of a postulated crack, structural integrity and function of the component would be maintained through the period of extended operation. As stated in Appendix E of WCAP-17096-NP, it would require a network of connected cracks to allow significant degradation of the upper core plate functionality. For the lower support forging/casting, the reduction in inspection requirements is further supported by the fact that the expansion from the UFW also triggers an EVT-1 inspection of the most likely part of the lower support forging/casting to experience cracking—the core barrel LFW.

   Based on these considerations of the degradation that could lead to an impact on the functionality of these two components, the reduced inspection coverage is justified.

c. If surface-breaking indications are discovered in the inspection of the initial 25% coverage of the upper core plate or the lower support forging/casting, the examination would be expanded to include the remaining accessible surfaces of the component. Table 4-6 and Table 5-3 of MRP-227, Revision 1 will be updated as shown below to reflect this change.

Table 4-6 of MRP-227, Revision 1 will be updated as shown below, including the addition of Note 3 to the table.

Expansion Item	Applicability	Effect （Mechanism）	Primary Link （Note 1）	Examination Method/Frequency （Note 1）	Examination Coverage
Lower Internals Assembly W3.4.Lower support forging or castings	All plants	Cracking （SCC） Aging Management （TE in Casting） Ref.[34]	W3.Upper Core Barrel Flange Weld （UFW）	Visual （VT-3） examination. Re-inspection every 10 years following initial inspection.	Minimum of 25% of bottom （non-core side） surface. （Note 3） See Figure 4-20.

Expansion Item	**Applicability**	**Effect (Mechanism)**	**Primary Link (Note 1)**	**Examination Method/Frequency (Note 1)**	**Examination Coverage**
Upper Internals Assembly W4.1.Upper core plate	All plants	Cracking （Fatigue）, Wear, Aging Management （IE）	W4.Lower Girth Weld （LGW）	Visual （VT-3） examination. Re-inspection every 10 years following initial inspection.	Minimum of 25% of core side surfaces. （Note 3） See Figure 4-19.

3. The stated minimum coverage requirement is the minimum if no significant indications are found. However, the Examination Acceptance criteria in Section 5 require that additional coverage must be achieved in the same outage if significant flaws are found. This contingency should be considered for inspection planning purposes.

Table 5-3 of MRP-227, Revision 1 will be updated as shown below to reflect the expansion in inspection coverage of these components should degradation be identified in the initial exam.

Primary Item	Applicability	Examination Acceptance Criteria (Note 1)	Expansion Link(s)	Expansion Criteria	Expansion Item Examination Acceptance Criteria
Core Barrel Assembly Upper flange weld (UFW)	All plants	Periodic enhanced visual (EVT-1) examination. The specific relevant condition is a detectable crack-like surface indication.	Upper girth weld (UGW) Lower flange weld (LFW) (Note 2) Upper axial welds (UAW) Lower support forging/casting	a. The confirmed detection of a surface breaking linear indication in the UFW shall require that inspection coverage of the UFW shall be extended to include 100% of the accessible length during the same outage. b. The confirmed detection and sizing of a surface-breaking indication with a length greater than two inches in the UFW shall require that the inspection be expanded to include the UGW and LFW by the completion of the next refueling outage. c. The confirmed detection and sizing of a surface breaking indication with a length greater than two inches in either the UGW or LFW shall require that the inspection be expanded to include the UAW by the completion of the next refueling outage. d. The confirmed detection of a surface-breaking indication with a length greater than two inches in the LFW shall require the inspection of the lower support forging or casting (25% of the non-core side surface) within the next three refueling outages. If an indication is found in this inspection of the lower support forging or casting, the examination coverage shall be expanded to 100% of the accessible surface of the non-core side surface of the lower support forging or casting during the same refueling outage.	The specific relevant condition for the expansion core barrel welds (UGW, LFW, UAW) and lower support forging or casting examinations is a detectable crack-like surface indication.

Core Barrel Assembly Lower girth weld (LGW)	All plants	Periodic enhanced visual (EVT-1) examination. The specific relevant condition is a detectable crack-like surface indication.	Upper core plate Lower support column bodies (cast and non-cast) Middle axial welds (MAW) Lower axial welds (LAW)	a. The confirmed detection and sizing of a surface-breaking indication with a length greater than two inches in the LGW shall require inspection of the upper core plate (25% of the core-side surface) within the next three refueling outages. If an indication is found in this inspection of the upper core plate, the examination coverage shall be expanded to 100% of the accessible surface of the core-side surface of the upper core plate during the same refueling outage. b. The confirmed detection and sizing of a surface-breaking indication with a length greater than two inches in the LGW shall require inspection of the lower support column bodies (cast and non-cast) within the next three refueling outages. c. The confirmed detection and sizing of a surface-breaking indication with a length greater than two inches in the LGW shall require that the inspections be expanded to include the lower core barrel cylinder axial welds by the completion of the next refueling outage.	a. The specific relevant conditions for the inspection of the upper core plate are broken or missing parts of the plate. b. The specific relevant conditions for the inspection of the lower support column bodies (cast and non-cast) are fractured, misaligned, or missing columns. c. The specific relevant condition for the expansion MAW and LAW inspections is a detectable crack-like surface indication.

3.5 Response to NRC RAI 15

3.5.1 NRC Question

Applicant/licensee Action Item (A/LAI) 3 was included in the NRC staff's final safety evaluation of MRP-227, Rev. 0 (Ref. 8) because MRP-227-A did not provide adequate guidance for applicants/licensees to document the specifics of the plant-specific existing programs in plant-specific RVI programs. With respect to Westinghouse existing plant-specific programs, MRP-227-A stated that "The guidance for guide tube support pins (split pins) is limited to plant-specific recommendations and thus have no generic reference. Subsequent performance monitoring should follow the supplier recommendations. Thus, they are not included in Table 4-9. The owner should review their specific design, upgrade status, and asset management plans for guide tube support pins (split pins)."

For Westinghouse split pins, similar guidance is included in MRP-227, Rev. 1 to that in MRP-227-A. The revised guidance states, "Additionally, in Westinghouse –design plants, the originally installed alloy X750 guide tube support pins (split pins) have been typically replaced with components with improved designs and less susceptible materials. The plant owner should review their specific design, upgrade status, and asset management plans for guide tube support pins (split pins). Thus the guide tube support pins (split pins) are not included in Table 4-9." However, the guidance is not sufficient because it does not specify that an applicant or licensee must include the specifics of the aging management program for split pins in its plant-specific RVI program. Also, the revised wording appears to imply that aging management is only necessary for Alloy X-750 split pins. Further, in Table 3-3, the material for the guide tube support pins is listed as Alloy X-750, implying that Type 316 split pins are a no additional measures component.

Therefore, the NRC staff requests that EPRI:

a. Clarify if type 316 stainless steel split pins require a plant-specific aging management program, or whether they are a "no additional measures component." Modify the wording of section 4.4 of MRP-227, Rev. 1 as necessary.

b. Discuss whether it would be appropriate to include a requirement in MRP-227, Rev. 1 that the specific aging management program for split pins be documented in the plant-specific RVI program, including the replacement and/or inspection schedule, replacement material, examination method and coverage, technical basis for the replacement schedule or the remaining life of the split pins (if already replaced), and technical basis for the inspection schedule or lack of inspections.

3.5.2 Industry Response

a. Control rod guide tube split pins (also called support pins) fabricated from Type 316 stainless steel are a "no additional measures component." This is based on the results of the failure modes, effects, and criticality analysis documented in MRP-191 where the split pins fabricated from

Type 316 stainless steel were assigned to Category A, while split pins fabricated from Alloy X-750 were assigned to Category C. These categorization results were based on material susceptibility to aging-related degradation and operating experience. Category A was assigned to components for which the aging effects are below the screening criteria or for which aging degradation significance is minimal. Further, the degradation of the split pin component is not a safety issue and is more of a concern for asset management. Therefore, the stainless steel 316 split pins are considered a "no additional measures component" and do not require a plant-specific aging management program.

The following two paragraphs from Section 4.4 of MRP-227, Revision 1 will be revised as follows in order to make this classification clear.

"A p~~P~~lant-specific Westinghouse-design item that ~~are already~~is managed within the plant program ~~or by modification include the~~is guidance for flux thimble tubes ~~and split pins~~. The bottom mounted instrumentation flux thimble tubes (applicable to all plants) are inspected to ensure pressure boundary integrity, as initially prompted by Bulletin IEB 88-09.

~~Additionally, in Westinghouse-design plants, the originally installed alloy X750 guide tube support pins (split pins) have been typically replaced with components with improved designs and less susceptible materials. The plant owner should review their specific design, upgrade status, and asset management plans for guide tube support pins (split pins). Thus, the guide tube support pins (split pins) are not included in Table 4-9."~~

Section 4.5 of MRP-227, Revision 1 is then updated as follows:

"It has been determined that no additional aging management is necessary for components in this group. Certain components designated as not requiring aging management may still be subject to ASME Code Section XI IWB-2500[2] in-service inspection requirements. In no case does a determination of no inspection for aging management in this guideline relieve utilities of any ASME Code Section XI IWB-2500 in-service inspection requirement unless specific relief is granted as allowed by 10 CFR 50.55a [4].

A plant-specific Westinghouse-design item that has been managed by modification is the upper internals guide tube support pins (split pins). The degradation of split pins is not a safety issue, but it is an asset management concern. In Westinghouse-design plants, the originally installed alloy X750 split pins have been typically replaced with components with improved designs and less susceptible materials, such as Type 316 stainless steel. Type 316 stainless steel split pins are a Category A component, which is assigned to components for which the aging effects are below the screening criteria or for which aging degradation significance is minimal. Therefore, the stainless steel 316 split pins are considered a "no additional measures component" and do not require a plant-specific aging management program. For this reason, the Type 316 stainless steel split pins are not included in Table 4-9.

The plant owner should review their specific design, upgrade status, and asset management plans for guide tube split pins. If the originally installed X750 split pins have not been replaced by components with a less susceptible material, a plant specific evaluation shall be performed to

determine an appropriate aging management program for the component and shall be included in the plant-specific RVI program for the period of extended operation, or until replacement is performed."

b. It is considered unnecessary to include a requirement for the plant-specific aging management program for split pins fabricated from Type 316 stainless steel to be documented in the plant-specific RVI program. This is based on the response to part A of this question and the categorization of those pins as a "no additional measures component."

The majority of Westinghouse plants with Alloy X-750 split pins (Category C from MRP-191) have replaced the pins with a material less susceptible to degradation. The few remaining plants that have X-750 split pins are covered by the text in Section 4.4 of MRP-227, Revision 1, which states that the "plant owner should review their specific design, upgrade status, and asset management plans for guide tube support pins (split pins)." It is good practice for plants to include reference to the split pin replacement in their RVI program as a means for meeting GALL requirements for evaluating and addressing industry operating experience and best practices with respect to its RVI aging management program. For X750 split pins, a statement is added to MRP-227, Revision 1 that the component shall be included in the plant-specific RVI program for the period of extended operation or until replacement is performed. The proposed update to Section 4.5 of MRP-227, Revision 1 is included in the response to part A above.

3.6 Response to NRC RAI 19

3.6.1 NRC Question

In Reference 9, Westinghouse submitted a notification pursuant to Title 10 of the Code of Federal Regulations (10 CFR) Part 21, that notified the NRC of a potential significant safety hazard due to guide card wear in four Westinghouse units that use ion nitride rod cluster control assemblies (RCCAs) in conjunction with 17x17 A or 17 x 17 AS style guide tubes. Guide card wear in these plants may occur more rapidly than predicted by WCAP-17451-P, Revision 1, "Reactor Internals Guide Tube Wear - Westinghouse Domestic Fleet Operational Projects," (Ref. 10) which is referenced in MRP-227, Rev. 1 with respect to the examination schedule, method, and coverage for CRGT guide plates (guide cards) in Westinghouse-design RVI. The NRC staff was also informed of a Westinghouse Nuclear Safety Advisory Letter (NSAL), issued in January 2017 (currently non-public), which addressed the accelerated wear issue and defined an accelerated schedule for the baseline guide card wear measurements at the four affected units.

The NRC staff requests that EPRI discuss how MRP-227, Rev. 1 and/or WCAP-17451-P, Rev. 1 should be modified to address the OE discussed in the 10 CFR Part 21 notification related to guide cards (Ref. 9).

3.6.2 Industry Response

MRP-227, Revision 1 references WCAP-17451-P, Revision 1 [9] for the guide card primary component inspection requirements. A PWROG program is currently in progress to evaluate the effect of the most recent operating experience from guide card inspections (as documented in NSAL-17-1, Revision 0 [8]) and to revise WCAP-17451-P, Revision 1 as appropriate. The next revision of WCAP-17451-P will be provided to the NRC for information, when available, in order to address the 10 CFR 21 notification. The reference to WCAP-17451-P within MRP-227, Revision 1 will not include the revision number and will state that the “latest NRC-reviewed or approved version” is applicable. Table 4-3 of MRP-227, Revision 1 will be updated as shown below, including modification to Note 7 to the table.

Primary Item	Applicability	Effect (Mechanism)	Expansion Link (Note 1)	Examination Method/Frequency (Note 1)	Examination Coverage
W1.Control Rod Guide Tube Assembly Guide plates (cards)	All plants	Loss of Material (Wear)	None	Per the requirements of WCAP-17451-P, including subsequent examinations (Note 7).	Examination coverage per the requirements of WCAP-17451-P (Note 7) See Figure 4-11

7. In WCAP-17451-P Revision 1 the baseline examination schedule has been adjusted for various CRGT designs, the extent of individual CRGT examination modified, and flexible subsequent examination regimens correlating to initial baseline sample size, accuracy of wear estimation and examination results. Initial inspection prior to the license renewal period may be required. Refer to the latest revision of WCAP-17451-P including the results of the NEI 03-08 Generic Topical Report screening and/or NRC review for the specific guidance elements.

3.7 Response to NRC RAI 20

3.7.1 NRC Question

For a number of Primary and Expansion weld items in Tables 4-2, 4-3, 4-5, and 4-6, the revised examination coverage in MRP-227, Rev. 1 specifies a percentage of the weld length or circumference "and adjacent base metal" shall be examined. The weld items are listed in Table 2 below. The NRC staff requests EPRI define what extent of the adjacent base metal must be examined (e.g., a certain distance from the weld fusion line or centerline).

Table 2 – Weld Items with Adjacent Base Metal to be Examined

Table	Item
4.2.	C5. Core Support Barrel Assembly Upper Flange Weld (UFW)
4.2	C6. Core Support Barrel Assembly – Middle Girth Weld (MGW)
4.3	W3. Core Barrel Assembly – Upper flange weld (UFW)
4.3	W4. Core Barrel Assembly – Lower Girth Weld (LGW)
4.5	C5.1 Core Support Barrel Assembly – Lower Girth Weld (LGW)
4.5	Core Support Barrel Assembly C5.2 Upper Girth Weld (UGW) C5.3 Upper Axial Weld (UAW)
4.5	Lower Support Structure C5.4 Lower Core Support Beams
4.5	Core Support Barrel Assembly C6.1. Middle Axial Weld(MAW) C6.2. Lower Axial Weld
4.5	Core Shroud Assembly (Welded) C2.1. Remaining Axial Welds
4.5	Core Shroud Assembly (Welded) C3.1. Remaining axial welds
4.6	Control Rod Guide Tube Assembly – W2.1.Remaining CRGT lower flange welds
4.6	Core Barrel Assembly – W3.1.Upper Girth Weld (UGW)
4.6	Core Barrel Assembly – W3.2.Upper Axial Weld (UAW)
4.6	Core Barrel Assembly – W3.3.Lower Flange Weld (LFW)
4.6	Core Barrel Assembly – W4.2. Middle Axial Welds (MAW) W4.3. Lower Axial Welds (LAW)

3.7.2 Industry Response

EPRI has defined the extent of adjacent based metal to examine under the NEI 03-08 protocol via implementation of the Inspection Standard Pressurized Water Reactor Internals (MRP-228 Revision 2).

MRP-228 Revision 2 Section 4.2 contains the following "Needed" requirement for Examination Systems:

Needed: All examination personnel, equipment, examinations, classification and measurement of indications, and documentation associated with visual examinations shall meet the requirements of Sections 2.3.4, 2.3.5, 2.3.6, 2.3.7, 2.3.8, and 2.3.9, respectively.

Section 2.3.6.4 "Area(s) Of Interest" states:

For EVT-1 or VT-1 examination of components, the area of interest shall consist of all accessible surfaces of the component. For welds, when adjacent base metal is specified in the examination coverage requirement, it is intended to include the base metal heat-affected zone adjacent to the weld. This is generally considered to be the entire width of the weld and 0.75 in. (19.1 mm) of the adjacent base material on each side of the weld. However, for small welds such as the control rod guide tube upper flange welds, the entire weld and 0.25 in. (6 mm) of the adjacent base material is the area of interest. To determine the adequacy of coverage, the area of interest may be further refined based on the damage mechanisms for the items identified in MRP-227. Such refined determinations of coverage adequacy shall be documented in the inspection report, as prepared or reviewed by a Level III visual examiner qualified to this standard, and shall be approved by the utility.

In Section 4.2.2 of MRP-227 Revision 1, the following text defines what extent of the adjacent base metal must be examined:

"When adjacent base metal is specified in the inspection coverage requirement, it is intended to include the base metal heat affected zone adjacent to the weld. If not otherwise specified, three quarter inch of base metal coverage may be assumed."

Industry acknowledges that this could be interpreted as a permissive not a requirement. Thus, to be consistent with the requirements in MRP-228 Section 2.3.6.4 the final staff-reviewed version of MRP-227 Revision 1, Section 4.2.2 will be slightly modified as follows:

"When the adjacent base metal is specified in the inspection coverage requirement, it is intended to include the base metal heat affected zone adjacent to the weld. ~~If not otherwise specified, three quarter inch of base metal coverage may be assumed.~~ The adjacent base metal to be examined is defined in Section 2.3.6.4 of MRP-228 and is referenced in the Section 4 Primary and Expansion component tables in this document."

In order to make this clear, the requirement to inspect ¾" of the adjacent base metal will be added to Tables 4-2 through 4-6 in the final staff-reviewed version of MRP-227, Rev. 1 for the welds to which the requirement applies. The table below shows the welds where ¾" adjacent base metal inspection is required.

MRP-227, R1 Table No.	MRP-227, R1 Component ID & Description	MRP-227, R1 Table No.	MRP-227, R1 Component ID & Description
Table 4-2	C2: Core Shroud Plate-Former Plate Weld	Table 4-5	C5.4: Lower Core Support Beams
Table 4-2	C3: Shroud Plates	Table 4-5	C6.1: Middle Axial Weld
Table 4-2	C5: Upper Flange Weld	Table 4-5	C6.2: Lower Axial Weld
Table 4-2	C6: Middle Girth Weld	Table 4-5	C2.1: Remaining Axial Welds
Table 4-2	C12: Deep Beams	Table 4-5	C3.1: Remaining Axial Welds
Table 4-3	W3: Upper Flange Weld	Table 4-6	W3.1: Upper Girth Weld
Table 4-3	W4: Lower Girth Weld	Table 4-6	W3.2: Upper Axial Weld
Table 4-5	C5.1: Lower Girth Weld	Table 4-6	W3.3: Lower Flange Weld
Table 4-5	C5.2: Upper Girth Weld	Table 4-6	W4.2: Middle Axial Weld
Table 4-5	C5.3: Upper Axial Weld	Table 4-6	W4.3: Lower Axial Weld

It should be noted that the base metal examination coverage for Item W2.1 in Table 4-6 was unintentionally omitted. This item should have the same base metal examination coverage as item W2 in Table 4-3: 0.25-inch of the base metal adjacent to the lower flange welds on the individual remaining CRGT assemblies.

The changes to be made to Table 4-6 in MRP-227, Revision 1 are shown below in red.

Expansion Item	Applicability	Effect (Mechanism)	Primary Link	Examination Method/Frequency	Examination Coverage
Control Rod Guide Tube Assembly W2.1.Remaining CRGT lower flange welds	All plants	Cracking (SCC, Fatigue) Aging Management (IE and TE)	W2.CRGT Lower Flange Welds	Enhanced visual (EVT-1) examination to determine the presence of crack-like surface flaws in flange welds. Subsequent examination on a ten-year interval.	A minimum of 75% of the CRGT assembly lower flange weld surfaces and 0.25-inch of the adjacent base metal for the flange welds not inspected under the primary link.

3.8 Response to NRC RAI 23

3.8.1 NRC Question

In Table 5-2, "CE Plants Examination Acceptance and Expansion Criteria," for the Core Shroud Assembly (welded) – Assembly, the examination acceptance criteria in MRP-227, Rev. 1 specifies a VT-3 examination but a VT-1 examination is specified in Table 4-2 for this item. MRP-227-A specified VT-1 in both tables for this item. Clarify whether VT-1 or VT-3 is the intended technique. If VT-3 is the intended technique, explain why this technique is acceptable to address the amount of physical separation expected if distortion is occurring.

3.8.2 Industry Response

The examination acceptance criteria in Table 5-2 for the Core Shroud Assembly (Welded) Assembly should specify a VT-1 examination, consistent with that specified in Table 4-2. This is also the same as what was required in MRP-227-A. Table 5-2 of MRP-227, Revision 1 will be updated.

The changes to be made to Table 5-2 in MRP-227, Revision 1 are shown below in red.

Primary Item	Applicability	Examination Acceptance Criteria	Expansion Link(s)	Expansion Criteria	Expansion Item Examination Acceptance Criteria
Core Shroud Assembly (Welded) Assembly	Plant designs with core shrouds assembled in two vertical sections	Visual (~~VT-3~~ VT-1) examination. The specific relevant condition is evidence of physical separation between the upper and lower core shroud sections.	None.	N/A	N/A

3.9 Response NRC RAI 24

3.9.1 NRC Question

In MRP-227, Rev. 1, Table 5-2, the expansion criteria for the UFW requires inspection of the UGW, LGW, and UAW by the completion of the next refueling outage. However, the lower core support beams require inspection within the next three refueling outages. In MRP-227-A, for the corresponding item in Table 5-2, the Core Support Barrel Assembly – Upper (core support barrel) flange weld, the expansion to the lower core support beams was required by the completion of the next refueling outage. What is the technical basis for changing the time frame for the expansion inspection of the lower core support beams to within the next three refueling cycles?

3.9.2 Industry Response

The change in the timing for the expansion to the lower core support beams from C5. Core Support Barrel Assembly – Upper Flange Weld (UFW) was based on two considerations.

The first consideration was an evaluation of the likelihood and consequence of aging-related degradation in the components. The lower core support beams are a redundant item, since there are multiple beams and multiple structural welds on those beams. This redundancy reduces the likelihood of having enough cracking to lose functionality. Per MRP-191, the consequence of degradation in the core support beams is low, as compared to the high consequence of core support barrel weld degradation, because of this difference in redundancy.

The second consideration supporting the need for a longer expansion inspection interval is the time required to develop an inspection technique. The core barrel weld expansions from the UFW can be inspected using a similar technique and tooling already developed for the UFW or other weld inspections. The lower core support beams are much more difficult to access and tooling has not been developed to perform an EVT-1 inspection of their welds. Given the relative difference in consequence and likelihood between the lower core support beams and core support barrel welds, it was considered prudent to allow more time for tooling and method development, supporting safety and ALARA for the inspection.

3.10 Response to NRC RAI 26

3.10.1 NRC Question

For the Westinghouse core barrel assembly, two welds have been reclassified from Primary to Expansion in MRP-227, Rev. 1. The nomenclature has also been changed for some of the welds and some of the weld items in MRP-227-A have been subdivided in MRP-227, Rev. 1. Table 3 below provides the MRP-227-A item name, and the equivalent MRP-227, Rev. 1 item name.

Table 3 – Westinghouse Core Barrel Assembly Weld Items Reclassified from Primary to Expansion

Table	MRP-227-A Primary Item	MRP-227, Rev. 1 Primary Item(s)	MRP-227, Rev. 1 Expansion Item(s)
4.3	Core Barrel Assembly – Upper and Lower Core Barrel Cylinder Girth Welds	Core Barrel Assembly W4. Lower Girth Weld (LGW) (Primary)	Core Barrel Assembly W3.1.Upper Girth Weld (expansion) (UGW)
4.3	Core Barrel Assembly – Lower Core Barrel Flange Weld	n/a	Core Barrel Assembly W3.3.Lower Flange Weld (LFW)

For Westinghouse, in MRP-227-A, Table 4-3, "Westinghouse Plants Primary Items," the upper and lower core barrel cylinder girth welds are Primary components for cracking due to SCC, IASCC, and fatigue. In MRP-227, Rev. 1, Table 4-3, "Westinghouse Plants Primary Items," the original item has been subdivided into two new items, the lower girth weld (LGW) and the upper girth weld (UGW). Only the LGW Is Primary in MRP-227, Rev. 1, while the UGW has been changed to an Expansion item. In addition, the equivalent component to the Core Barrel Assembly – Lower Core Barrel Flange Weld in MRP-227, Rev. 1, the Core Barrel Assembly W3.3.Lower Flange Weld, has also been reclassified from Primary to Expansion. The NRC staff notes that the CE item equivalent to the LFW is C7.Core Support Barrel Assembly – CSB Flexure Weld (CSBFW), which remains a Primary item in MRP-227, Rev. 1.

In addition, per Table 5-3, the expansion to Table 4.6, Core Barrel Assembly W3.2.Upper Axial Weld (UAW), would only occur if indications are found in either the UGW or the LFW, which are also expansion items. Therefore, it could be as much as four years between the detection of degradation in the primary item until the UAW are examined.

The NRC staff requests that EPRI:

a. Justify reclassifying the UGW and LGW from Primary to Expansion.

b. Justify making the UAW a "secondary expansion" to the UGW and LFW.

c. Justify reclassifying the LFW from Primary to Expansion. Explain why the LFW classification is not consistent with the analogous CE component, the CSBFW, which is classified as Primary.

3.10.2 Industry Response

a. In this response, it was assumed that the question was only asking about the reclassification of the UGW from primary to expansion, since the LGW remains as a primary component in MRP-227, Revision 1.

   During preparation for inspections, more details on the typical naming used for the core barrel welds and more precise locations for the welds were found. This information on naming and location provided the basis for the renamed core barrel weld components listed in MRP-227, Revision 1 and the specific aging-related degradation mechanisms assigned to each weld. This was also an input to the assignment of welds to be primary or expansion components.

   These clarifications resulted in several changes, as noted in the question. The upper and lower core barrel cylinder girth welds were revised to provide more detail in MRP-227, Revision 1, resulting in the LGW being subject to SCC, IASCC, IE, and fatigue, and the UGW being subject to SCC.

   During the review of MRP-227, Revision 1 and its basis documents to respond to this question, it was noted that the degradation mechanisms assigned to the LGW were incorrect. Fatigue should not be assigned as a screened in mechanism. The appropriate degradation mechanisms are determined by an expert panel and are documented in MRP-191 (Revision 0 and Revision 1), which is then reflected in Table 3-3 of MRP-227-A and MRP-227, Revision 1. Based on the original screening and expert panel results, the LGW is only subject to SCC, IASCC, and IE. The error appears to have occurred when transferring information from MRP-191 to MRP-227 and was present in MRP-227-A and MRP-227, Revision 1. This will be corrected by deleting "fatigue" from Table 4-3 for inclusion in the final staff-reviewed version of MRP-227, Rev. 1 as shown in the markup at the end of this response.

   Provided with this revised information on applicable degradation mechanisms and with the updated nomenclature for each of the welds, the core barrel welds were assigned a more logical structure in MRP-227, Revision 1 using its sampling and lead component strategies. The UGW was moved to be an Expansion component from the UFW because both welds have similar low normal operating stresses and were screened in for the same degradation mechanism (SCC), and the UFW has the added potential of elevated bending stresses due to the proximity of the upper flange. If the sampling inspection of the UFW as a Primary lead component detects evidence of cracking degradation, the UGW will require expansion inspection according to the requirements of MRP-227, Revision 1 to monitor for further occurrence of the degradation mechanism.

b. The UAW was assigned as a "secondary expansion" component to the UGW and LFW based on both the likelihood and consequence of degradation in the axial welds as compared to the girth welds. The UAW is not as severely loaded as a girth weld, so it is less susceptible to cracking degradation mechanisms. Cracking of the UAW would not result in a loss of core support; therefore, the consequence of failure would be lower than that of a girth weld. This logic forms the basis for the UAW being made a "secondary expansion" from the UGW and LFW.

c. Similar to the discussion of the UGW in the response to part A of this question, the LFW was moved to be an Expansion component from the UFW due to the location experiencing lower stresses. Thus, the UFW is expected to lead the LFW in experiencing SCC degradation. If the

sampling inspection of the UFW as a Primary lead component detects evidence of cracking degradation, the LFW will require expansion inspection according to the requirements of MRP-227, Revision 1 to monitor for further occurrence of the degradation mechanism.

The core support barrel flexure weld (CSBFW) in certain CE plant designs is analogous to the LFW in Westinghouse plant designs in terms of the general location on the core barrel; however, the two components are quite different in design. CSBFW is a smaller weld that attaches the CE lower support structure to a flexure on the lower flange of the core barrel. This flexure is intended to accommodate relative thermal expansion in the two component assemblies. Given the large difference in geometry and design, the two welds are susceptible to different degradation mechanisms. The CSBFW is subject to both SCC and fatigue. The LFW is only susceptible to SCC (see part A of this response for a correction to the applicable degradation mechanisms in the LFW). The differences in design and the differences in applicable degradation mechanisms result in the CSBFW being a Primary item while the LFW is reclassified as an Expansion component.

Table 4-3 of MRP-227, Revision 1 is updated as follows:

W4.Core Barrel Assembly Lower girth weld (LGW)	All plants	Cracking (SCC, IASCC~~, Fatigue~~), Aging Management (IE)	Upper core plate, Lower support column bodies (cast, non-cast), middle axial welds (MAW), lower axial welds (LAW)	Enhanced visual (EVT-1) examination, no later than 2 refueling outages from the beginning of the license renewal period and subsequent examination on a ten-year interval.	A minimum of 25% of the OD circumference of the LGW and adjacent base metal shall be examined. (Notes 8 and 10) See Figure 4-13.

3.11 Response to NRC RAI 27

3.11.1 NRC Question

The MRP-227, Revision 1 report includes Section 7.3, “Reactor Internals Guidelines Inspection Requirement.” In this section, EPRI states “[e]ach commercial U.S. PWR unit shall implement the requirements of Tables 4-1 through 4-9 and Tables 5-1 through 5-3 for the applicable design.”

Section 7.3 in MRP-227, Revision 1, omitted the following information that was previously included in Section 7.3 of the MRP-227-A report:

> Consistent with the requirements of NEI 03-08, if the guidance contained in Table 4-1 through 4-9 and/or Tables 5-1 through 5-3 cannot, need not, or will not be implemented as written, a technical justification must be prepared that clearly states what requirement cannot, need not, or will not be met and why; what alternative action is being taken to satisfy the objective or intent of the guidance; and, why the alternative action is acceptable. Examples of alternatives that may be justifiable are: elevation of an Expansion component to Primary; substitution of an equivalent or more rigorous examination than is required by the tables; or destructive testing in lieu of nondestructive examination, such as the case where one or more of the primary components is being replaced. Since the Expansion components are also “needed” requirements, the technical justification for not fully implementing a Primary component examination or not implementing it in a manner consistent with its intent, would be expected to include disposition of the associated Expansion components.
>
> When submittal of a deviation from work products or elements is required, the justification shall be reviewed and approved in accordance with the applicable plant procedures with the additional responsibility for deviation from a ‘Needed’ element that an internal independent review is performed and that concurrence is obtained from the responsible utility executive. Further, as stipulated in the Implementation Protocol (Appendix B) of NEI 03-08, a utility is required to notify the Issue Program (e.g., the MRP) and the NRC.

Justify the basis for omitting these paragraphs from the scope of Section 7.3 of the MRP-227, Revision 1 report.

3.11.2 Industry Response

The two aforementioned paragraphs in Section 7.3 of MRP-227-A were removed from the same section of MRP-227, Rev. 1 to reduce redundancy within the document.

MRP-227 contains Mandatory and Needed Requirements under the EPRI MRP (Materials Reliability Program) Issue Program as stated in Section 7.1 of MRP-227, Revision 1. The Reactor Internals Guidelines Implementation Requirement (Section 7.3) contains the “Needed” requirement under NEI 03-

08 to implement Tables 4-1 through 4-9 and 5-1 through 5-3 for the applicable plant design. This means that NEI-03-08 requirements apply to the implementation of this document, including Appendix B, Section 8 of NEI-03-08 on Deviations.

Appendix B, Section 8 of NEI 03-08 outlines the protocol for utility processing of deviations from Mandatory or Needed requirements. Appendix B, Section 8.1.c. states that “if at any time a utility does not implement any ‘Mandatory’ or ‘Needed” elements of an approved guideline, the utility shall notify the NRC.”

Additionally, the details provided in the above paragraphs are not considered to be the only possible approaches for dealing with the need for a technical justification. It was not the intention of Section 7.3 of MRP-227 to be prescriptive in how technical justifications should be approached.

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ATTACHMENT 2 – Industry Responses to RAIs Related to B&W-Design PWRs

(Ref.AREVA report ANP-3610)

ABSTRACT

By letter dated December 21, 2015, the Electric Power Research Institute (EPRI) submitted for U. S. Nuclear Regulatory Commission (US NRC) staff review Topical Report MRP-227, Revision 1, “Materials Reliability Program: Pressurized Water Reactor Internals Inspection and Evaluations Guidelines.” The US NRC has issued Requests for Additional Information (RAIs) on this submittal. This report provides the AREVA Inc. responses for RAIs 1, 2, 3, 4, 6, 11, 13, 17, 18, 21, 22, 25, and 27.

1.0 INTRODUCTION AND SUMMARY

By letter dated December 21, 2015, the Electric Power Research Institute (EPRI) submitted for U. S. Nuclear Regulatory Commission (US NRC) staff review Topical Report MRP-227, Revision 1, “Materials Reliability Program: Pressurized Water Reactor Internals Inspection and Evaluations Guidelines” (Reference 1). The US NRC has issued Requests for Additional Information (RAIs) on this submittal (Reference 2) and this report provides the responses to those RAIs assigned to AREVA Inc. The responses to these RAIs are supported by technical information contained in MRP-189, Revision 2 (Reference 3) and MRP-231, Revision 3 (Reference 4).

Upon receipt of the RAIs, EPRI and AREVA Inc. reviewed the RAIs and determined who would respond to each RAI. The responses for RAIs 1, 2, 3, 4, 6, 11, 17, 18, 21, 22, and 25 were assigned to AREVA Inc. The responses to RAIs 5, 7, 8, 9, 10, 12, 14, 15, 16, 19, 20, 23, 24, and 26 were assigned to Westinghouse by EPRI. The responses to RAIs 13 and 27 are shared by AREVA Inc. and Westinghouse. For completeness, this document contains sections for all twenty-seven RAIs; however, the responses for those RAIs that were assigned to Westinghouse only say “Assigned to Westinghouse by EPRI by the Division of Responsibility.”

2.0 REQUESTS FOR ADDITIONAL INFORMATION

The NRC RAIs are addressed, as noted in Section 1.0, in Section 2.1 through Section 2.27.

2.1 *RAI 1*

2.1.1 Statement of RAI 1

In MRP-227, Revision 1 (Reference 1), for the following Babcock & Wilcox (B&W) primary components, the schedule for the initial (baseline) examination changed from "during the next 10-year ISI [inservice inspection]" to "during the next 10-year ISI interval."

- B2. Control Rod Guide Tube (CRGT) Assembly - spacer castings
- B3. Vent Valve Assembly
  - o A. Vent valve top retaining ring
  - o B. Vent valve bottom retaining ring
- B10. Core Barrel Assembly-Baffle plates
- B11. Core Barrel Assembly - Locking devices, including locking welds, of baffle-to-Former bolts and internal baffle-to-baffle-bolts

Clarify what this means; for example, does "during the next 10-year ISI" mean during the next scheduled 10-year ISI examination of the reactor vessel internals (RVI), or does it mean sometime during the next 10-year ISI interval? If the latter, does that mean these examinations may not be performed until up to 20 years from now, if the current 10-year ISI interval started today? If this is the case, justify waiting up to 20 years to perform the baseline examination.

2.1.2 Response to RAI 1

The updated wording in MRP-227 Revision 1 (Reference 1) does not allow initial examinations 20 years into the period of extended operation. Per Section 4 on page 4-2 of MRP-227, Revision 1, the term "10-year ISI interval" is intended to mean the plant's existing schedule associated with removal of the core barrel, i.e., during the next scheduled 10-year ISI interval examination. As defined in the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel (B&PV) Code Section XI (Reference 5), the first inspection interval is defined as 10 years following the initial start of plant commercial startup, and successive inspection intervals are defined as 10 years following the previous inspection interval. Each inspection interval may be extended by as much as 1 year and may be reduced without restriction, provided the examinations required for the interval have been completed. Successive intervals shall not extend more than 1 year beyond the original pattern of 10 year intervals and shall not exceed 11 years in length. Therefore, for the baseline (i.e., initial) examinations, the intention of this wording is for examinations to be performed prior to the end of the fourth ASME ISI interval and not more than 11 years since the previous ASME ISI interval was completed, i.e., what is allowed by Section XI of the ASME B&PV Code. This is also consistent with the stipulations stated in Section 4.2.6 of MRP-227, Revision 1 for subsequent examination intervals.

2.2 *RAI 2*

2.2.1 Statement of RAI 2

For MRP-227-A item in Table 4-1, "B&W Plants Primary Components:"

<u>Plenum Cover Assembly & Core Support Shield Assembly</u>

Plenum cover weldment rib pads

Plenum cover support flange

Core Support Shield (CSS) top flange

The revised item description in MRP-227, Revision 1, Table 4-1 is:

"B1.Plenum Cover Assembly & Core Support Shield Assembly"

a. Plenum cover weldment rib pads

b. Plenum cover support flange

c. Plenum cover support ring

d. CSS top flange"

The examination coverage changed from "Determination of differential height of top of plenum rib pads to reactor vessel seating surface, with plenum in reactor vessel" to "Determination of differential height of top of plenum rib pads/plenum cover support ring location to reactor vessel seating surface, with plenum in reactor vessel."

The change to this item was to add the plenum cover support ring as a subcomponent and to add this subcomponent as an additional reference point for the physical measurement.

The plenum cover support ring appears to be a new subcomponent added in MRP-227, Revision 1. The plenum cover support ring is addressed in MRP-189, Revision 1, “Screening, Categorization, and Ranking of B&W-Designed PWR Internals Component Items” (Reference 6[2]), and was determined to be Category A. The plenum cover assembly – weldment rib pads and plenum cover assembly – support flange were determined to be Category C for wear in MRP-189. Therefore, the NRC staff requests that the Electric Power Research Institute (EPRI) clarify why the plenum cover support ring was added as a subcomponent and how and why the support ring was added as a reference location for making the physical measurements.

2.2.2 Response to RAI 2

During preparation of MRP-189, Revision 2 (Reference 3), the preparers determined that the plenum cover support ring, in addition to the plenum rib pads, is in contact with the reactor vessel closure head and that additional wording clarification should be included. Core clamping measurements have been performed consistently at all B&W-designed units and have included the plenum cover support ring. Therefore, the text in MRP-189, Revision 2 has been updated to reflect this change.

[2] “Reference 6” is a direct quotation from the NRC RAI letter (Reference 2). The same reference herein is Reference 7.

The plenum cover support ring and the plenum rib pads are machined to a common plane (see Figure 2-3 and accompanying text in Section 2.3.1 of MRP-189, Revision 2). Figure 2-1 (Reference 6) provides a sketch of the profile view of the core clamping details. Figure 2-2 (Reference 6) provides a sketch showing the interference height measurement. In both figures, the solid vertical blue box depicts the plenum cover support ring and the horizontal white box depicts a plenum rib pad. In Figure 2-1 only, the horizontal patterned box depicts the plenum cover support flange. The core clamping area is composed of three mating surfaces: the location where the core support shield upper flange sits on the reactor vessel ledge, the location where the plenum cover support flange sits on the upper core support shield flange, and (of interest to this RAI) the area where the common plane of the plenum rib pads and top of the plenum cover support ring interface with the reactor vessel closure head. The actual clamping occurs when the reactor vessel closure head is positioned on the reactor vessel and the studs are tensioned. The stud loading is principally resisted by the interface between the reactor vessel and the reactor vessel closure head. The force transmitted through the internals between the reactor vessel closure head flange and the reactor vessel flange is referred to as the core clamping force. Therefore, in addition to MRP-189, both MRP-231 and MRP-227 were updated accordingly.

Figure 2-1. Sketch Showing Core Clamping Components to be Measured (Reference 6)

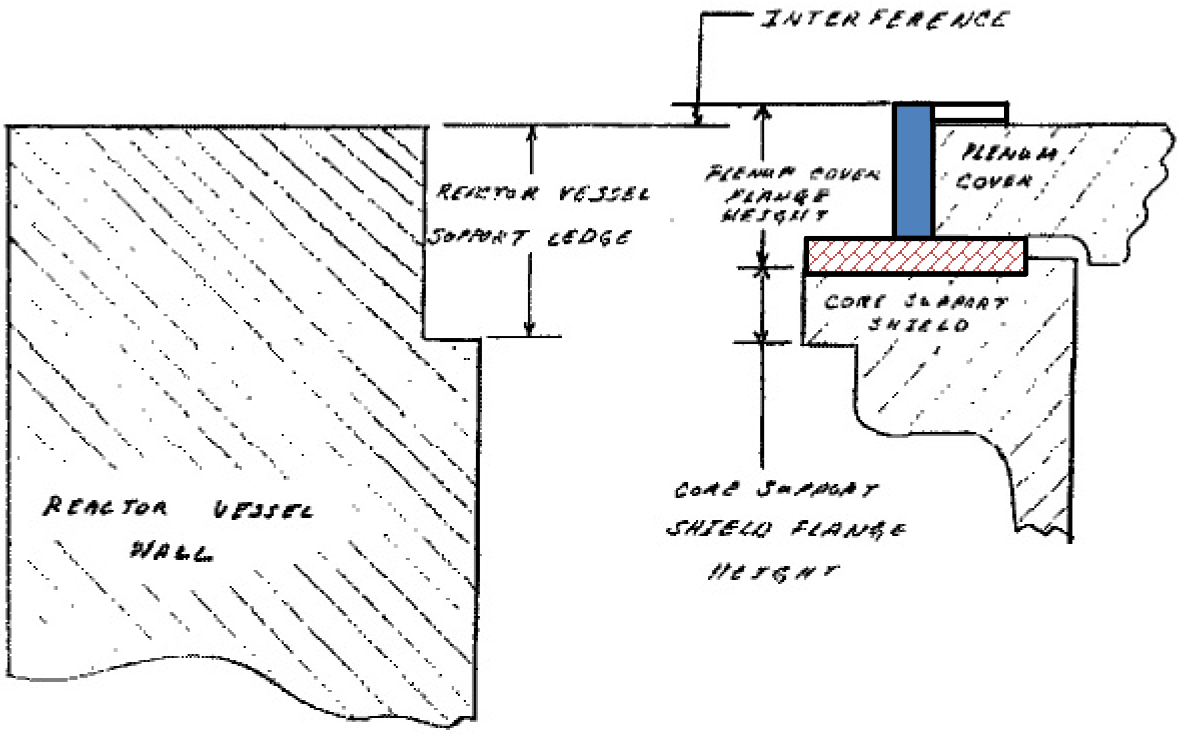
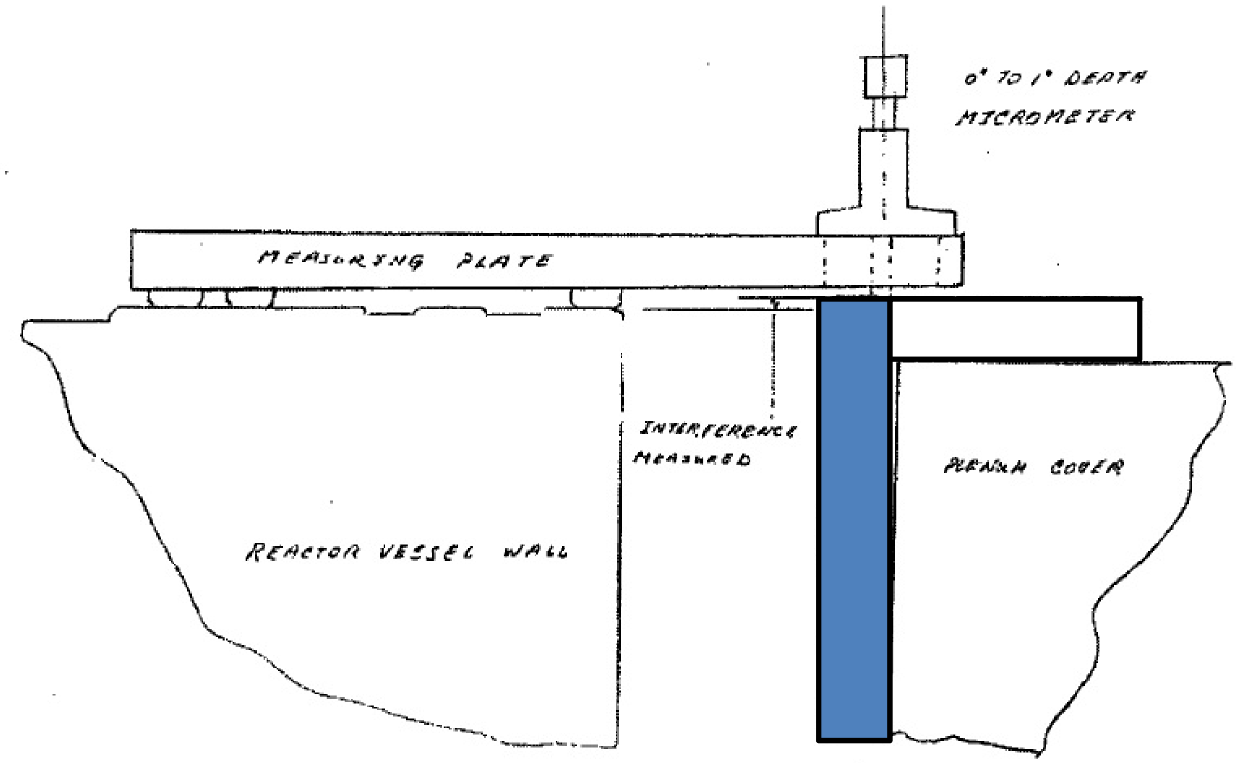


Figure 2-2. Sketch Showing Interference Height Measurement (Reference 6)



2.3 *RAI 3*

2.3.1 Statement of RAI 3

In Table 4-1, the “Control Rod Guide Tube Assembly – CRGT,” spacer castings previously had no expansion link. An expansion link to the vent valve bodies has now been added in MRP-227, Revision 1. The vent valve bodies were not an expansion component in MRP-227-A. According to MRP-189, Revision 1, the vent valve bodies are cast austenitic stainless steel (CASS), as are the CRGT spacer castings. Since the vent valve bodies were previously a no additional measures component, the U.S. Nuclear Regulatory Commission (NRC) staff requests that EPRI explain why the vent valve bodies were made an expansion component for the CRGT spacer castings.

2.3.2 Response to RAI 3

In the evaluation performed for originally installed vent valve bodies, all of the vent valve bodies were previously screened out (i.e., Category A for all aging degradation mechanisms of concern) in MRP-189, Revision 1 (Reference 7) due to their low ferrite content. However, as noted in Section 3.2, item J.2 (page 3-7) and in Table 3-2 Note 6 (page 3-15) of MRP-189, Revision 2 (Reference 3), each Babcock and Wilcox (B&W) utility was also supplied with spare vent valves in addition to the originally installed vent valves. It is now known that some spare vent valves have been installed, replacing the original vent valves. Currently, the certified material test reports (CMTRs) for spare vent valves are not readily available. Therefore, because the ferrite content of these replacement vent valve bodies is not currently known, the vent valve bodies are currently assumed to have a ferrite content exceeding 20%.

Per Section 3.2.6.2 of MRP-231, Revision 3 (Reference 4), the CRGT spacer castings are made of Type CF3M castings. The vent valve bodies are made of Type CF8 castings. The vent valve bodies and the CRGT spacer castings are located above the core and their operating conditions are similar, i.e., at hot leg temperature with an irradiation dose too low to cause irradiation embrittlement (IE). Hence, their extent of thermal aging embrittlement (TE) is expected to be similar. However, Type CF3M material contains 2% to 3% percent molybdenum, which may potentially contribute to a higher TE for the CRGT spacer castings than the Type CF8 vent valve bodies, depending on the casting method and ferrite content. Therefore, the CRGT spacer castings are categorized as Primary component items, because the potential degradation is expected to bound that of the vent valve bodies. The vent valve bodies are thus categorized as Expansion component items for the CRGT spacer castings.

Additionally, as remarked in Note 4 in Table 4-4 of MRP-227, Revision 1 (Reference 1), each utility may be able to identify the body serial number (S/N) and heat number for each installed vent valve during vent valve exercising. If the CMTR and chemical composition of the vent valve body are subsequently found and hence the ferrite content of the currently installed vent valve bodies can be verified to be under the TE screening criterion for the material, the vent valve bodies can be removed as an Expansion component item and default back to a No Additional Measures component item. As an alternative, each utility may be able to utilize a statistical argument for estimating the ferrite content of the currently installed vent valve bodies using the results obtained in PWROG-15032-NP (Reference 8). If, at any time, the ferrite content of any installed vent valve body is concluded to be greater than the screening criterion for TE in MRP-175 (Reference 9), that vent valve body will need to be included as an Expansion component item.

2.4 *RAI 4*

2.4.1 Statement of RAI 4

In Table 4-1, the schedule for the initial (baseline) ultrasonic (UT) examination of the Core Barrel Assembly - Baffle-to-former Bolts changed from “no later than two refueling outages from the beginning of the license renewal period” to “volumetric (UT) examination during the next 10-year ISI interval.” Since it is not clear when the next ten-year ISI interval starts (it could be up to ten years from the current date), this could result in the baseline examination being significantly later than MRP-227-A would require. It was not clear to the NRC staff whether this assumes all six operating B&W units have completed baseline UT examinations already. The NRC staff, therefore, requests EPRI provide the following information:

1. Does the initial baseline UT examination schedule for the baffle-to-former bolts in MRP-227, Revision 1 assume an examination of baffle-to-former bolts has been completed within two refueling outages from the beginning of the period of extended operation?

2. If not, justify the change in the schedule for the initial (baseline) UT examination of the baffle-to-former bolts.

2.4.2 Response to RAI 4

2.4.2.1 Part 1

Yes, the initial baseline UT examination schedule for the baffle-to-former (B-F) bolts in MRP-227, Revision 1 (Reference 1) assumes an examination of B-F bolts has been completed within two refueling outages from the beginning of the period of extended operation for each operating B&W unit.

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2.4.2.2 Part 2

A response to this part of this RAI is not required to be provided based on the response to Part 1 of this RAI.

2.5 *RAI 5*

Assigned to Westinghouse by EPRI by the Division of Responsibility

2.6 *RAI 6*

2.6.1 Statement of RAI 6

In Table 4-1, Item B11., "Core Barrel Assembly – Locking Devices," including locking welds, of baffle-to-former bolts and internal baffle-to-baffle bolts, has applicable aging mechanisms of irradiation assisted stress corrosion cracking (SCC) (IASCC), irradiation embrittlement (IE) including the detection of missing, non-functional, or removed locking devices or welds, and has as an Expansion link "locking devices, including locking welds, of the external baffle-to-baffle bolts and core barrel-to-former bolts." However, in MRP-227, Revision 1, a new Note 8 has been added for the expansion link, which states that "the aging degradation mechanism of IASCC is only applicable to the baffle-to-former bolt and internal baffle-to-baffle bolt locking devices, not the baffle-to-former bolt and internal baffle-to-baffle bolt locking device welds." However, under the expansion link column in Table 4-1, the expansion link for Item B11 is described as locking devices, including locking welds, of the external baffle-to-baffle bolts and core barrel-to-former bolts.

The NRC staff therefore requests the following information:

a) Clarify whether the expansion link column or Note 8 is correct.

b) If Note 8 is correct, explain why IASCC is not applicable to the locking device welds, and why there are no Expansion links for the welds.

2.6.2 Response to RAI 6

2.6.2.1 Part a

Note 8 in Table 4-1 and the Expansion link in MRP-227, Revision 1 (Reference 1) are both correct.

As discussed in Section 3.2.3 and stated in Note 1 of Table 3-3 in MRP-231, Revision 3 (Reference 4), the locking devices for the baffle-to-former (B-F) bolts and internal baffle-to-baffle (B-B) bolts are Primary component items for irradiated-assisted stress corrosion cracking (IASCC) with no Expansion link. Therefore, Note 8 of Table 4-1 in MRP-227, Revision 1 is correct as written. However, for clarification, the “Effect (Mechanism)” column of Table 4-1 for Item B11., “Core Barrel Assembly—Locking devices, including locking welds, of baffle-to-former bolts and internal baffle-to-baffle bolts” in MRP-227, Revision 1 will be revised as follows, with the reference notes remaining the same as those currently in MRP-227, Revision 1 Table 4-1:

Table 2-1. New Effect (Mechanism) Entry for Primary Core Barrel Assembly Locking Devices and Locking Welds in Table 4-1 of MRP-227, Revision 1

Primary Item	Effect (Mechanism)
B11.Core Barrel Assembly Locking devices, including locking welds, of baffle-to-former bolts and internal baffle-to-baffle bolts	Locking Devices: Cracking (IASCC, IE) including the detection of missing, non-functional, or removed locking devices (Note 3) Locking Welds: Cracking (IE) including the detection of missing, non-functional, or removed locking device welds (Note 3, 8)

2.6.2.2 Part b

As stated in Note 1 of Table 3-1 in MRP-189, Revision 2 (Reference 3), page B-7 of MRP-175 (Reference 9) suggests an IASCC fluence screening criterion of 2E21 n/cm^2, E > 1.0 MeV (3 dpa) for highly stressed component items such as bolts, springs, and multi-pass welds. As shown in Table 3-3 of MRP-189, Revision 2, the welds for the B-F bolt locking devices to the baffle plates (WC-15, WC-94), the welds for the internal B-B bolt locking devices to the baffle plates (WC-16, WC-17), the welds for the CB-F bolt locking devices to the core barrel cylinder (WC-14, WC-15), and the welds for the external B-B bolt locking devices to the baffle plates (WC-16, WC-134, WC-135) are either fillet, locking, or plug welds, not multi-pass welds. Therefore, the aging degradation mechanism of IASCC is screened out (i.e., Category A) for these locking device welds and there are no Primary/Expansion relationships for IASCC. As noted in Table 2-1, these welds are considered potentially susceptible to irradiation embrittlement (IE).

2.7 *RAI 7*

Assigned to Westinghouse by EPRI by the Division of Responsibility

2.8 *RAI 8*

Assigned to Westinghouse by EPRI by the Division of Responsibility

2.9 *RAI 9*

Assigned to Westinghouse by EPRI by the Division of Responsibility

2.10 *RAI 10*

Assigned to Westinghouse by EPRI by the Division of Responsibility

2.11 *RAI 11*

2.11.1 Statement of RAI 11

In Table 4-4, B&W Plants Expansion Components, Item B7.1 Core Barrel Assembly – Upper Thermal Shield (UTS) bolts and their locking devices and Item B7.2 Core Barrel Assembly - Surveillance specimen holder tube (SSHT) bolts and their locking devices, had changes to the “Effect (mechanism)” information. Specifically, irradiation creep/irradiation stress relaxation (IC/ISR)/Wear/Fatigue were added for the SSHT bolts.

Note 7 has also been added to Table 4-4 indicating that this table entry for the SSHT bolts also includes the aging degradation mechanisms of ISR/IC/wear/fatigue for the compression collar and washer for the SSHT bolt. The compression collars for the SSHT bolt are not included in the screening and failure mode, effects and criticality analysis (FMECA) documented in MRP-189, "Materials Reliability Program: Screening, Categorization, and Ranking of B&W-Designed PWR Internals Component Items” (Reference 6[3]) and MRP-190, "Materials Reliability Program: Failure Modes, Effects, and Criticality Analysis of B&W-Designed PWR Internals” (Reference 7[4]).

The NRC staff, therefore, requests EPRI:

a) Explain why the new aging mechanisms of IC/ISR/Wear/Fatigue was added for the SSHT bolts.

b) Clarify whether the compression collars were left out of the screening and FMECA process as an oversight, or whether the compression collars are the same as the SSHT bolt locking cups and tie plates that are included in the screening and FMECA. If the latter, explain why the screening and FMECA results for these components changed.

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[3] “Reference 6” is a direct quotation from the NRC RAI letter (Reference 2). The same reference herein is Reference 7.

[4] “Reference 7” is a direct quotation from the NRC RAI letter (Reference 2). The same reference herein is Reference 10.

2.11.2 Response to RAI 11

2.11.2.1 Part a

As noted in Table 3-1 in MRP-189, Revision 2 (Reference 3), the screening criteria for irradiation-enhanced stress relaxation and creep (ISR/IC) is a dose of greater than or equal to 1.3E20 n/cm^2, E > 1 MeV (0.2 dpa), applied to all bolted or spring locations. Additionally, wear and fatigue are applicable to bolted or spring items where ISR/IC is applicable. Table 3-2 in MRP-189, Revision 2 lists the dose at 54 EFPY for the surveillance specimen holder tube (SSHT) bolts as 1.5E21 n/cm^2, E > 1 MeV (note that the 48 EFPY fluence value listed in MRP-189, Revision 1 (Reference 7) is 7.60E19 n/cm^2, E > 1 MeV). Therefore, since the fluence value at 54 EFPY for the SSHT bolts exceeds the ISR/IC screening criteria, ISR/IC, wear, and fatigue are applicable to the SSHT bolts. As noted in Table 4-2 of MRP-189, Revision 2, the failure modes, effects, and criticality analysis (FMECA) categorized the susceptibility to ISR/IC/wear/fatigue of the SSHT bolts as Category “C” and the safety consequence as “1”, with a final categorization as Category “B” (note; this categorization accounts for stress corrosion cracking [SCC] in addition to ISR/IC/wear/fatigue). Therefore, the age-related degradation mechanisms of ISR/IC/ wear/fatigue of the SSHT bolts are included in MRP-227, Revision 1.

2.11.2.2 Part b

The only currently-operating B&W unit with functional SSHT bolts is Davis-Besse Unit 1 (DB-1). The original SSHT bolts were replaced at DB-1 in 1984 and 1990; a records search was completed in 2010, after the publication of MRP-189, Revision 1 (Reference 7) and MRP-190 (Reference 10), which identified that the design of these SSHT bolts at DB-1 utilizes a bolt, compression collar, spherical washer, and a tie-plate/crimp locking cup assembly. These additional components items, which were not known to exist during preparation of MRP-227-A, are now listed in Table 3-2 of MRP-189, Revision 2 (Reference 3) and screen as Category Not A for various aging degradation mechanisms, including ISR/IC/wear/fatigue for the compression collar. The revised FMECA documented in Table 4-2 of MRP-189, Revision 2 determined the compression collar to be Category B due to ISR/IC/wear/fatigue. As discussed in MRP-231, Revision 3 (Reference 4), Section 3.2.4, the SSHT bolts, including the compression collars, are Expansion component items in MRP-227, Revision 1 (Reference 1).

2.12 *RAI 12*

Assigned to Westinghouse by EPRI by the Division of Responsibility

2.13 *RAI 13*

2.13.1 Statement of RAI 13

The following four areas pertaining to Tables 4-1 through 4-3 could be informed by Operating Experience (OE) related to the actual accessibility achieved for primary system components during baseline inspections:

1. In MRP-227, Revision 1, has OE been used to modify or clarify examination coverage requirements of MRP-227-A based on the actual accessibility achieved during the examinations completed to date? If so, identify the components that have had examination coverage revised based on OE, and describe the reason for the change. If coverage requirements have not been revised based on OE, justify why this has not been done.

2. Has OE with actual coverage achieved resulted in any primary component that was previously considered to be accessible being reclassified as inaccessible, either because of the percentage of the component surface area, length, or population that is accessible was insufficient to provide reasonable assurance of functionality, or because insufficient coverage was achieved of the most likely portion of the component to exhibit degradation? Identify any primary components that have been reclassified as inaccessible and identify what alternate measures, such as an engineering analysis, were taken to provide reasonable assurance of component functionality.

3. For primary components reclassified as inaccessible, were the expansion links reevaluated for these components?

4. For any primary components reclassified as inaccessible, were alternate primary components selected?

2.13.2 Response to RAI 13

2.13.2.1 Part 1

Operating experience (OE) has not yet been used to modify or clarify examination coverage of MRP-227-A based on actual accessibility achieved during the examinations completed to date for the Babcock and Wilcox (B&W)-designed units. Coverage has not yet been revised based on OE because, except for one instance, the expected coverage (including allowances for minimum populations of bolts and locking devices) has been able to be achieved during each examination to date. At one US PWR, only about 99% coverage was achieved during the control rod guide tube (CRGT) spacer casting VT-3 examination; 10 spacer castings at each of the 4 screw locations were inaccessible for VT-3 examination due to permanent obstruction from the reactor vessel level monitoring system (RVLMS) installed at one CRGT location. This OE was not available during preparation of MRP-227, Revision 1 (Reference 2). As a result, the examination coverage in MRP-227, Revision 1 for this Table 4-1 entry will be modified as noted by underlined text in Table 2-2:

Table 2-2. New CRGT Spacer Casting Entry for Table 4-1 of MRP-227, Revision 1

Primary Item	Applicability	Effect (Mechanism)	Expansion Link (Note 1)	Examination Method/Frequency (Note 1, 2)	Examination Coverage
Control Rod Guide Assembly CRGT spacer castings	All plants	Cracking (TE), including the detection of fractured spacers or missing screws (Note 3)	Vent valve bodies	Visual (VT-3) examination during the next 10-year ISI interval. Subsequent examination prior to the end of each 10-year ISI interval.	Accessible surfaces at each of the four screw locations (at every 90º) of 100% of the CRGT spacer castings. (limited accessibility) <u>(Note 10)</u> See Figure 4-5.

Notes to Table 2-2:

1. Examination acceptance criteria and expansion criteria for the B&W component items and welds are detailed in Table 5-1.

2. Initial examinations may be scheduled concurrently with the next 10-year ISI but shall be performed no later than the final outage of the current 10-year ISI interval at the time of entry into the period of extended operation, except where otherwise noted.

3. Loss of ductility and fracture toughness, which are induced by thermal aging or neutron irradiation embrittlement, are not directly monitored. These effects are indirectly managed by performing periodic visual examinations capable of detecting cracking in the component.

10. <u>At one of the CRGT locations at ANO-1, a device was installed to monitor water level that required removal of the center control rod drive mechanism and installation of an assembly that blocks access to all of the CRGT spacer castings except the top spacer casting. Therefore, for all plants except ANO-1, 100% of the CRGT spacer castings refers to 690 CRGT spacer castings. There is no requirement to examine this set of CRGT spacer castings at this one CGRT assembly at ANO-1 and 100% is defined as 680 CRGT spacer castings for ANO-1.</u>

2.13.2.2 Part 2

Operating experience (OE) to-date has not resulted in the reclassification of a component as inaccessible for the Babcock and Wilcox (B&W)-designed units.

2.13.2.3 Part 3

A response to this part is not required based on the response to Part 2 of this RAI.

2.13.2.4 Part 4

A response to this part is not required based on the response to Part 2 of this RAI.

2.14 *RAI 14*

Assigned to Westinghouse by EPRI by the Division of Responsibility

2.15 *RAI 15*

Assigned to Westinghouse by EPRI by the Division of Responsibility

2.16 *RAI 16*

Assigned to Westinghouse by EPRI by the Division of Responsibility

2.17 *RAI 17*

2.17.1 Statement of RAI 17

For Table 4-4, B&W Plants Expansion Items, Core Barrel Assembly, B11.1.Locking Devices, including locking welds, of the external baffle-to-baffle bolts and core barrel-to-former bolts, the primary link changed from:

"…locking devices, including locking welds, of baffle-to-former bolts **or** internal baffle-to-baffle bolts," to

"B11.Locking devices, including locking welds, of baffle-to-former bolts **and** internal baffle-to-baffle bolts.

Does the change from “or” to “and” mean degradation now has to be exhibited in both the locking devices for the baffle-to-former bolts and the locking devices for the internal baffle-to-baffle bolts for the expansion to be required, whereas in MRP-227-A the expansion would be required if only one of these items exhibited degradation? If so, justify the changes.

2.17.2 Response to RAI 17

No, the change from “or” to “and” noted in this RAI does not mean degradation now has to be exhibited in both the locking devices for the baffle-to-former bolts and the locking devices for the internal baffle-to-baffle bolts for the Expansion to be required. The Primary link for the B11.1 Expansion items was, and still is, required to be both types of locking devices. This change was editorial in nature only. The same text is now consistently used in Tables 4-1, 4-4, and 5-1 of MRP-227, Revision 1 (Reference 1).

2.18 *RAI 18*

2.18.1 Statement of RAI 18

In Table 4-4, the Lower Grid Assembly – Item B10.3., “Lower Grid Rib Section,” has been added as an additional Expansion link for Primary Item B10., “Core Barrel Assembly – Baffle Plates. Lower Grid Assembly,” – Item B10.3., “Lower Grid Rib Section,” was not included in MRP-227-A as either a Primary or Expansion component. The NRC staff therefore requests EPRI explain why this item has apparently been recategorized from “no additional measures” to “expansion.”

2.18.2 Response to RAI 18

The following text is summarized in Table 2-3.

The lower grid rib section was screened as Category Not A for irradiation embrittlement (IE) in Table 3-2 of MRP-189, Revision 1 (Reference 7) and MRP-189, Revision 2 (Reference 3). The lower grid rib section was categorized as Category “A” as a part of the FMECA in MRP-189, Revision 1, owing the categorization to a susceptibility ranking of “B” and a safety consequence of “1”.

As detailed in Note 14 on page 4-13 of MRP-189, Revision 2, the safety consequence of failure of the lower grid rib section was increased from that assigned in MRP-190 (Reference 10) (safety consequence of “1” to a safety consequence of “2”) based on its direct core support function (the need to assign a safety consequence of “2”) and redundancy because of the need for multiple flaws to initiate and grow to critical flaw size in multiple ribs to cause a safety concern (the need to not assign a safety consequence of “3”). Additionally, MRP-189, Revision 2 introduced a new table (Table 4-1), not in MRP-189, Revision 1, detailing the susceptibility metrics for IE, irradiation-enhanced stress relaxation and creep (ISR/IC), and irradiation-assisted stress corrosion cracking (IASCC). Based on Table 4-1 of MRP-189, Revision 2, the aging degradation susceptibility metric of the lower grid rib section to IE is a “C” due to the expected 54 EFPY dose of 3.2E21 n/cm^2, E > 1 MeV listed in Table 3-2 of MRP-189, Revision 2. Therefore, the preliminary categorization was revised to Category “C” based on a susceptibility ranking of “C” and a safety consequence of “2”.

Section 3.2.3.1 of MRP-231, Revision 3 (Reference 4) describes the classification of the lower grid rib section as an Expansion item based on its expected dose as compared to the baffle plates; this is reflected in Table 3-1 of MRP-227, Revision 1 (Reference 1).

Table 2-3. Summary of Changes for the Lower Grid Rib Section between MRP-227-A and MRP-227, Revision 1

	MRP-189 Screening Result	MRP-189 FMECA Result	MRP-231 Aging Management Strategy
Lower Grid Rib Section in MRP-227-A	Category "Not A" for IE	Category "A" based on a susceptibility of "B" and safety consequence of "1"	Category "A" (not considered in MRP-231)
Lower Grid Rib Section in MRP-227, Revision 1	Category "Not A" for IE	Category "C" based on a susceptibility of "C" and safety consequence of "2"	Expansion (to Baffle Plates)

2.19 *RAI 19*

Assigned to Westinghouse by EPRI by the Division of Responsibility

2.20 *RAI 20*

Assigned to Westinghouse by EPRI by the Division of Responsibility

2.21 *RAI 21*

2.21.1 Statement of RAI 21

In Table 5-1, the Core Barrel Assembly – Baffle-to-former bolts expansion criteria have changed. In MRP-227-A, the Expansion criteria is “Confirmed unacceptable indication in greater than or equal to 5% (or 43) of the baffle-to-former bolts, provided that none of the unacceptable bolts are on former elevations 3, 4, and 5, or greater than 25% of the bolts on a single baffle plate, shall require an evaluation of the internal baffle-to-baffle bolts for the purpose of determining whether to examine or replace the internal baffle-to-baffle bolts. The evaluation may include external baffle-to-baffle bolts and core barrel-to-former bolts for the purpose of determining whether to replace them.” In MRP-227, Revision 1, the expansion criteria is “Confirmed unacceptable indications in greater than or equal to 5% of the baffle-to-former bolts (including previously failed/removed bolts) shall require an evaluation of the baffle-to-baffle bolts and the core barrel-to-former bolts by the completion of the next refueling outage. The evaluation shall also assess functionality of the core barrel assembly with aging degradation of the baffle-to-baffle bolts and core barrel-to-former bolts for the purpose of determining continued operation or replacement.”

The criteria requiring expansion if greater than 25 percent of the bolts on one baffle plate are degraded would result in expansion if clustering of degraded bolts was present, which has been seen in recent OE with baffle-former bolt degradation in Westinghouse-design RVI. It is also not clear why the language regarding bolts on former elevations 3, 4, and 5 has been removed from the expansion criteria.

The NRC staff therefore requests that EPRI provide the technical basis for the changes to the expansion criteria for the baffle-to-former bolts in B&W plants. The response should address the following items:

a) An explanation for the removal of the language from the expansion criteria related to bolts on former levels 3, 4, and 5, and whether this results in less conservativism. If less conservative, provide a justification for the reduction in conservatism.

b) Why was the expansion criterion of more than 25% of the bolts on a single plate [degraded] removed in Revision 1 especially considering recent OE with clustered baffle-former bolt degradation?

2.21.2 Response to RAI 21

2.21.2.1 Part a

The Expansion criteria were updated to only include consideration that an active age-related degradation mechanism in the baffle-to-former (B-F) bolts would be present, as aging degradation drives the Expansion inspections. A structural analysis performed for the operating Babcock and Wilcox (B&W) units demonstrated that failure of B-F bolts on all rows other than rows 3, 4, and 5 was an acceptable configuration for continued operation, as that configuration maintained fuel impact assessment boundary conditions. Additional details regarding this structural analysis are in Sections 3.2.2.1.3 and 3.2.2.1.5 of MRP-231, Revision 3 (Reference 4). Since the presence of these B-F bolts is a consideration for continued operation, and not Expansion, associated language was removed from the Expansion criteria related to the B-F bolts. As this language is not associated with potential core barrel-to-former (CB-F) bolt, internal baffle-to-baffle (B-B) bolt, or external B-B bolt failures as predicted by B-F bolt failures, removal of the language does not result in less conservatism.

2.21.2.2 Part b

The exact mechanism(s) of irradiation-assisted stress corrosion cracking (IASCC) is not fully understood but the current consensus is that IASCC results from the synergistic effects of irradiation damage to the material, the aggressive water environment, and the stress state. The water environment is similar for all B&W B-F bolts. The accumulated dose at 40-60 years may not exceed the industry-accepted IASCC screening criteria of 3 dpa (Reference 9) for some of these austenitic stainless steel B-F bolts, but there still may be some irradiation-enhanced stress relaxation and creep (ISR/IC), thus decreasing the stresses for these B-F bolts. For B-F bolts with very high dose, ISR/IC could decrease the stress such that little stress is remaining in the B-F bolt. Therefore, irradiation damage and stress state are potential competing interests for IASCC initiation.

The compilation of all recent operating experience (OE) with clustered B-F degradation being observed at some Westinghouse-designed units was not yet available during the preparation of MRP-227, Revision 1. However, this OE has been assessed and it was determined that several key stress drivers for IASCC appear to exist at the Westinghouse-designed units that do not exist at the B&W-designed units (Reference 11). These are: 1) higher applied stresses due to the pressure differential inherent in the downflow configuration at the applicable Westinghouse-designed units (B&W units have always been an upflow configuration), and 2) bolting installation and design characteristics that appear to be more susceptible to cracking (i.e., bolt length, initial bolt torque levels, the fabrication process associated with some internal hex head designs, and potentially higher stress concentration factors in the head-to-shank area). While sufficient information is currently not available to AREVA Inc. to completely quantify these stress drivers, less IASCC initiation is expected in the B&W units. This is supported by the current OE for the B&W units (one B-F bolt with a UT indication in over 3,400 B-F bolts examined), which has a high likelihood of being a random failure and not an indication of an active degradation mechanism having initiated (Reference 11).

A cluster of failed B-F bolts is thought to be the result of a random distribution of degraded B-F bolts (by IASCC) near to each other where the degraded B-F bolts lose their ability to carry their expected load and consequently nearby or neighboring B-F bolts carry increased loads, leading to adjacent failures (Reference 12), which is more likely to occur with a downflow configuration.

The requirement for greater than 25% of the B-F bolts on a single baffle plate to require Expansion to the CB-F, internal B-B, and external B-B bolts would indicate that clustering of IASCC initiation would be expected. This type of clustering is not expected with the B&W units since higher applied stresses due to failure of neighboring B-F bolts from a downflow configuration are not inherent to the B&W design due to the lower pressure differential; therefore, such an Expansion criterion is not necessary.

2.22 *RAI 22*

2.22.1 Statement of RAI 22

In Table 5-1, the examination acceptance criteria and expansion criteria for the Core Barrel Assembly – Baffle Plates have changed. In MRP-227-A, the examination acceptance criteria column in Table 5-1 stated that the specific relevant condition is readily detectable cracking in the baffle plates. In MRP-227, Revision 1, this has been changed to state the specific relevant condition is readily detectable cracking connecting openings in the baffle plates (i.e., each bolt hole and flow hole).

With respect to expansion criteria, in MRP-227-A, the expansion criteria states;

> Confirmed cracking in multiple (2 or more) locations in the baffle plates shall require expansion, with continued operation of former plates and the core barrel cylinder justified by evaluation or by replacement by the completion of the next refueling outage.

In MRP-227, Revision 1, the expansion criteria state:

> Gross cracking (if confirmed) within one inch of a bolt or flow hole location in the baffle plates shall require:
>
> a) An evaluation of the former plates and the core barrel cylinder for the purpose of determining continued operation or repair/replacement by the completion of the next refueling outage. Alternatively, repair/replacement activities may be initiated based on results of a best effort former plate and core barrel cylinder examination.
>
> b) That the Visual (VT-3) examination be expanded by the completion of the next refueling outage to include 100% of the accessible portions of the lower grid rib section heat-affected zones adjacent to the in-core monitoring instrumentation (IMI) guide tube spider-to-lower grid rib section welds.

The relevant condition now requires cracking connecting openings in baffle plates, rather than just detectable cracking. Also, the expansion criteria in MRP-227, Revision 1 seem inconsistent with the relevant conditions since the relevant conditions require linkage of openings by cracking, while the expansion criteria only seem to require cracking within one inch of an opening.

The NRC staff therefore requests the following information:

a) Provide a technical justification for the change in the definition of the relevant condition for the baffle plates, specifically, the new requirement that the cracking link openings in the baffle plates.

b) Provide a technical justification for the change in the expansion criteria for the baffle plates.

c) Clarify whether expansion is only required if cracking links two or more openings or whether expansion would be required if cracking is present within one inch of any opening.

2.22.2 Response to RAI 22

2.22.2.1 Part a

The relevant condition from Table 5-1 of MRP-227-A will be retained in MRP-227, Revision 1 (Reference 1). This is reflected by changes (strikethrough of text) shown in the "Primary Item Examination Acceptance Criteria" column in Table 2-4 in this document.

2.22.2.2 Part b

The Expansion criteria from Table 5-1 of MRP-227, Revision 1 (Reference 1) were updated for two reasons: 1) previously, MRP-227-A required two instances of confirmed cracking before Expansion; MRP-227, Revision 1 only requires one instance of confirmed gross cracking (within one inch of a bolt or flow hole, which is the examination coverage for both MRP-227-A and MRP-227, Revision 1), and 2) the Expansion criteria were updated to reflect Licensee/Applicant Action Item 6 from MRP-227-A (part a of the Expansion criteria) and the lower grid rib section as an Expansion item to the baffle plates (see response to RAI 18 in Section 2.18.2 of this document, part b of the Expansion criteria).

Table 2-4. New Baffle Plate Entry for Table 5-1 of MRP-227, Revision 1

Primary Item	Applicability	Primary Item Examination Acceptance Criteria (Notes 1,2)	Expansion Link(s)	Expansion Criteria	Expansion Item Examination Acceptance Criteria
Core Barrel Assembly Baffle Plates	All plants	Visual (VT-3) examination The specific relevant condition is readily detectable cracking ~~connecting openings~~ in the baffle plates ~~(i.e., each bolt hole and flow hole).~~	Former plates Core barrel cylinder (including vertical and circumferential seam welds) Lower grid rib section	Gross cracking (if confirmed) within one inch of a bolt or flow hole location in the baffle plate shall require: a) An evaluation of the former plates and the core barrel cylinder for the purpose of determining continued operation or repair/replacement by the completion of the next refueling outage. Alternatively, repair/replacement activities may be initiated based on results of a best effort former plate and core barrel cylinder examination. b) That the VT-3 examination be expanded by the completion of the next refueling outage to include 100% of the accessible portions of the lower grid rib section heat-affected zones adjacent to the IMI guide tube spider-to-lower grid rib section welds.	N/A

Notes to Table 2-4:

4. The examination acceptance criterion for visual examination is the absence of the specific relevant condition(s).

5. Refer to MRP-231 (Reference 4) Section 3.3 for additional details on examination acceptance and expansion criteria.

2.22.2.3 Part c

As stated in Table 2-4, expansion is required if there is confirmed gross cracking within one inch of a bolt or flow hole location in the baffle plate.

2.23 *RAI 23*

Assigned to Westinghouse by EPRI by the Division of Responsibility

2.24 *RAI 24*

Assigned to Westinghouse by EPRI by the Division of Responsibility

2.25 *RAI 25*

2.25.1 Statement of RAI 25

In Table 4-1, or Item B15. “IMI Guide Tube Assembly Spiders and Spider welds,” – the examination coverage changed from “100% of top surfaces of 52 spider castings and welds to the adjacent lower grid rib section” in MRP-227-A to “Spiders: 100% of the accessible top surfaces and 100% of the accessible spider surfaces adjacent to the spider casting welds” and “Spider welds: 100% of the accessible welds to the adjacent lower grid rib section.”

The NRC staff requests that EPRI explain why the description of the examination coverage for this item changed, and explain the significance of this change.

2.25.2 Response to RAI 25

The description of the examination coverage was updated to clarify the examination coverage by separating the incore monitoring instrumentation (IMI) guide tube spiders and the welds to identify the specific areas of concern. The addition of the words “accessible” throughout the examination coverage description for these items have been added to maintain consistency with the examination coverage descriptions of other components items and welds in MRP-227, Revision 1 (Reference 1).

2.26 *RAI 26*

Assigned to Westinghouse by EPRI by the Division of Responsibility

2.27 *RAI 27*

2.27.1 Statement of RAI 27

The MRP-227, Revision 1 report includes Section 7.3, “Reactor Internals Guidelines Inspection Requirement.” In this section, EPRI states “[e]ach commercial U.S. PWR unit shall implement the requirements of Tables 4-1 through 4-9 and Tables 5-1 through 5-3 for the applicable design.”

Section 7.3 in MRP-227, Revision 1, omitted the following information that was previously included in Section 7.3 of the MRP-227-A report:

> Consistent with the requirements of Nuclear Energy Institute (NEI) document NEI 03-08 (Reference 13), if the guidance contained in Table 4-1 through 4-9 and/or Tables 5-1 through 5-3 cannot, need not, or will not be implemented as written, a technical justification must be prepared that clearly states what requirement cannot, need not, or will not be met and why; what alternative action is being taken to satisfy the objective or intent of the guidance; and, why the alternative action is acceptable. Examples of alternatives that may be justifiable are: elevation of an Expansion component to Primary; substitution of an equivalent or more rigorous examination than is required by the tables; or destructive testing in lieu of nondestructive examination, such as the case where one or more of the primary components is being replaced. Since the Expansion components are also “needed” requirements, the technical justification for not fully implementing a Primary component examination or not implementing it in a manner consistent with its intent, would be expected to include disposition of the associated Expansion components.

When submittal of a deviation from work products or elements is required, the justification shall be reviewed and approved in accordance with the applicable plant procedures with the additional responsibility for deviation from a ‘Needed’ element that an internal independent review is performed and that concurrence is obtained from the responsible utility executive. Further, as stipulated in the Implementation Protocol (Appendix B) of NEI 03-08, a utility is required to notify the Issue Program (e.g., the EPRI Materials Reliability Program, EPRI-MRP) and the NRC.

Justify the basis for omitting these paragraphs from the scope of Section 7.3 of the MRP-227, Revision 1 report.

2.27.2 Response to RAI 27

This text was omitted because key parts of the previous text, such as the requirement to provide a technical justification for a deviation from a Needed or Mandatory Requirement or the contents of the NEI 03-08 (Reference 13) Implementation Protocol, are included by reference to NEI 03-08.

3.0 REFERENCES

1. Letter, (via email) David Czufin (Chairman, PMMP Executive Committee) and Anne Demma (EPRI, MRP Program Manager) to Joe Holonich, (NRC), “Report Transmittal: Materials Reliability Program: Pressurized Water Reactor Internals Inspection and Evaluations Guideline (MRP-227 Revision 1), EPRI, Palo Alto, CA, 2015, 3002005349. Ref: EPRI Project Number 689,” MRP 2015-040, December 21, 2015, NRC Accession Number ML15358A046.

2. Letter, Joseph J. Holonich, (NRC Senior Project Manager) to Brian Burgos, (MRP Program Manager), “Request for Additional Information for Electric Power Research Institute Topical Report MRP-227, Revision 1, ‘Materials Reliability Program: Pressurized Water Reactor Internals Inspection and Evaluations Guideline’,” (CAC NO. MF7740), May 15, 2017, NRC Accession Number ML17079A027.

3. Materials Reliability Program: Screening, Categorization, and Ranking of Babcock & Wilcox-Designed Pressurized Water Reactor Internals Component Items and Welds (MRP-189, Revision 2). EPRI, Palo Alto, CA: 2014. 3002004238.

4. Materials Reliability Program: Aging Management Strategies for B&W Pressurized Water Reactor Internals (MRP-231-Revision 3). EPRI, Palo Alto, CA: 2014. 3002004284.

5. American Society of Mechanical Engineering, Boiler and Pressure Vessel Code, Section XI, Division 1, “Rules for Inservice Inspection of Nuclear Power Plant Components,” American Society of Mechanical Engineers, New York, NY, 2001 Edition, Plus 2003 Addenda, or later.

6. Letter, Timothy G. Wells (Southern Nuclear Operating Company) to U. S. Nuclear Regulatory Commission Document Control Desk, EPRI Letter MRP 2012-029, “Revised Responses to NRC Request for Additional Information (RAI) on PWR Owners Group (PWROG) WCAP-17096, Revision 2, ‘Reactor Internals Acceptance Criteria Methodology and Date Requirements,’ (TAC No. ME4200) PA-MSC-0473,” June 14, 2012, NRC Accession Number ML12171A374.

7. Materials Reliability Program: Screening, Categorization, and Ranking of B&W-Designed PWR Internals Component Items (MRP-189-Rev. 1). EPRI, Palo Alto, CA: 2009. 1018292.

8. PWROG-15032-NP, Revision 0, “PA-MSC-1288 Statistical Assessment of PWR RV Internals CASS Materials,” NRC Accession Numbers ML16068A245 and 16068A246.

9. Materials Reliability Program: PWR Internals Material Aging Degradation Mechanism Screening and Threshold Values (MRP-175). EPRI, Palo Alto, CA: 2005. 1012081.

10. Materials Reliability Program: Failure Modes, Effects, and Criticality Analysis of B&W-Designed PWR Internals (MRP-190). EPRI, Palo Alto, CA: 2006. 1013233.

11. AREVA Inc. Customer Service Bulletin 16-02, July 14, 2016, NRC Accession Number ML16215A061.

12. Westinghouse NSAL 16-01, Revision 1, August 1, 2016, NRC Accession Number ML16222A513.

13. Guidelines for the Management of Materials Issues, NEI 03-08, Nuclear Energy Institute, Washington DC, Latest Edition.

Date: January 30, 2018

To: Document Control Desk
Attn: Joe Holonich
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555-001

From: Mike Hoehn II, Ameren Missouri, MRP Integration Committee Chairman
Brian Burgos, EPRI, MRP Program Manager

SUBJECT: RESPONSES TO NRC REQUEST FOR ADDITIONAL INFORMATION FOR ELECTRIC POWER RESEARCH INSTITUTE TOPICAL REPORT MRP-227, REVISION 1, "MATERIALS RELIABILITY PROGRAM: PRESSURIZED WATER REACTOR INTERNALS INSPECTION AND EVALUATIONS GUIDELINE" (CAC NO. MF7740)

Dear Sir:

This letter transmits the industry's responses to the NRC requests for additional information (RAI) issued in reference 1 related to MRP-227, Revision 1 (reference 2), entitled "Materials Reliability Program: Pressurized Water Reactor Internals Inspection and Evaluations Guideline." EPRI received the formal RAIs from the NRC in a letter dated 5/15/2017 [ADAMS accession number ML17079A027]. EPRI discussed preliminary responses to the RAIs with NRC staff during public meetings on 7/12-13/2017 [ML17159A432], 9/6/2017 [ML17248A542], and 10/5/2017 [ML17278A034].

Responses to the following RAIs are included in this transmittal: 5, 9, 10, 12 and 16. Responses to the other RAIs were provided in a previous transmittal in reference 3. Enclosed are four (4) copies of this letter and the attachments.

The responses provided in this transmittal include recommended changes to MRP-227, Revision 1 which are considered changes to the guidance, as opposed to just clarifications. As such, the responses and recommended markups herein were reviewed and endorsed by the EPRI Executive Oversight Committee (PMMP) since the recommended changes affect the Section 4 and 5 tables of MRP-227, Revision 1, which are identified as an NEI-03-08 "Needed" requirements.

If you have additional questions or require further information, please contact Kyle Amberge (kamberge@epri.com, (704) 595-2039) or Brian Burgos (bburgos@epri.com, (724) 610-8559) or Myself (mhoehn@ameren.com, (314) 225-1543).

Sincerely,

Mike Hoehn II, Ameren-Missouri
MRP Integration Committee Chair

Brian Burgos, Program Manager
Materials Reliability Program

References:

1. U.S. Nuclear Regulatory Commission Letter, "Request for Additional Information for Electric Power Research Institute Topical Report MRP-227, Revision 1, 'Materials Reliability Program: Pressurized Water Reactor Internals Inspection and Evaluation Guideline' (CAC No. MF7740)," dated May 15, 2017 (ADAMS Accession No. ML17079A027).
2. Materials Reliability Program: Pressurized Water Reactor Internals Inspection and Evaluation Guidelines (MRP-227, Revision 1). EPRI, Palo Alto, CA: 2015. 3002005349.
3. EPRI letter MRP 2017-027, dated 10/16/2017

Project No. 669
cc: Jeff Poehler, NRC-NRR

ATTACHMENT 1

Responses to NRC Requests for Additional Information (RAIs) on MRP-227, Revision 1

1.0 Background and Introduction

This report provides responses to the U.S. Nuclear Regulatory Commission (NRC) Requests for Additional Information (RAIs) [1] related to the staff's review [2] of MRP-227, Revision 1 [3].

2.0 References

1. U.S. Nuclear Regulatory Commission Letter, "Request for Additional Information for Electric Power Research Institute Topical Report MRP-227, Revision 1, 'Materials Reliability Program: Pressurized Water Reactor Internals Inspection and Evaluation Guideline' (CAC No. MF7740)" May 15, 2017 (ADAMS Accession No. ML17079A027).
2. U.S. Nuclear Regulatory Commission Letter, "Acceptance Review of Electric Power Research Institute Topical Report MRP-227, Revision 1, 'Materials Reliability Program: Pressurized Water Reactor Internals Inspection and Evaluations Guideline,' (TAC No. MF7740)," July 15, 2016 (ADAMS Accession No. ML16154A063).
3. *Materials Reliability Program: Pressurized Water Reactor Internals Inspection and Evaluation Guidelines (MRP-227, Revision 1)*. EPRI, Palo Alto, CA: 2015. 3002005349.
4. EPRI - Report Transmittal: Materials Reliability Program: Pressurized Water Reactor Internals Inspection and Evaluations Guideline (MRP-227, Revision 1), December 21, 2015 (ADAMS Accession No. ML15358A046)
5. WCAP-17096-NP-A, Rev 2, "Reactor Internals Acceptance Criteria Methodology and Data Requirements." August 31, 2016 (ADAMS Accession No. ML16279A320)
6. Revision 1 to the Final Safety Evaluation of Electric Power Research Institute (EPRI) Report, Materials Reliability Program (MRP) Report 1016596 (MRP-227), Revision 0, "Pressurized Water Reactor (PWR) Internals Inspection and Evaluation Guidelines" (TAC No. ME0680), December 16, 2011 (ADAMS Accession No. Ml11308A770).
7. *Materials Reliability Program: Pressurized Water Reactor Issue Management Table, PWR-IMT Consequence of Failure (MRP-156, Revision 0)*. EPRI, Palo Alto, CA: 2005. 1012110.
8. Pressurized Water Reactor Owners Group Report, PWROG-14048-P, Rev. 2, "Functionality Analysis: Lower Support Columns," August 2017.
9. *EPRI Report, Pressurized Water Reactor Primary Water Chemistry Guidelines, Revision 7*. EPRI, Palo Alto, CA: 2014. 3002000505,

10. PWROG Letter, OG-17-62, "Submittal of PWROG-14048-P, Revision 1, 'Functionality Analysis: Lower Support Columns,' to the NRC for Information Only (PA-MSC-1103)," March 1, 2017 (ADAMS Accession No. ML17066A266).
11. *Materials Reliability Program: PWR Internals Material Aging Degradation Mechanism Screening and Threshold Values (MRP-175, Revision 0)*. EPRI, Palo Alto, CA: 2005. 1012081.
12. *Materials Reliability Program: Inspection Standard for Pressurized Water Reactor Internals – 2015 Update (MRP-228, Rev. 2)*. EPRI, Palo Alto, CA: 2015. 3002005386
13. Pressurized Water Reactor Owners Group Report, PWROG-15032-NP, Rev. 0, "PA-MSC-1288 Statistical Assessment of PWR RV Internals CASS Materials," November 2015.
14. NRC Staff Assessment of PWROG-15032, "Office of Nuclear Regulations Staff Assessment of the Pressurized Water Reactor Owner's Group Report PWROG-15032-NP, Revision 0, 'PA-MSC-1288 Statistical Assessment of PWR RV Internals CASS Materials," (ADAMS Accession No. ML16250A001).
15. NRC Staff Assessment of PWROG-14048, Rev. 1, "U.S. Nuclear Regulatory Commission Staff Assessment of PWROG-P, Rev. 1, 'Functionality Analysis: Lower Support Columns,'" (ADAMS Accession No. ML17251A905).
16. Westinghouse document WCAP-9251, Revision 0, "Scram Deflection Test Report 17x17 Guide Tubes, 96 Inch, and 150 Inch," December 1977. (Westinghouse Proprietary)

3.0 RAI Responses

Responses to individual NRC RAIs are provided in this section. Each section contains the RAI exactly as transmitted by the NRC, followed by the proposed response. Responses to RAIs 1-4, 6, 11, 17-18, 21-22, and 25 are not provided, as they are specific to B&W plants.

3.1 Response to NRC RAI 5

3.1.1 NRC Question

The required examination coverage in Table 4-2, "CE Plants Primary Components," and Table 4-3, "Westinghouse Primary Components," for four weld items, all of which are classified as high-consequence components in MRP-191, "Materials Reliability Program: Screening, Categorization, and Ranking of Reactor Internals Components for Westinghouse and Combustion Engineering PWR Design (MRP-191)," has been changed from 100 percent of the accessible surfaces of the [weld]" in MRP-227-A, to essentially a 25 percent sample of the weld circumference in MRP-227, Rev. 1. Table 1 below lists the old and new component item designations and the revised coverage requirement in MRP-227, Rev. 1.

Table 1 – Combustion Engineering and Westinghouse Core Support Barrel/Core Barrel Welds with Coverage Reduction in MRP-227, Rev. 1

MRP-227-A Item	Equivalent MRP-227, Rev. 1 Item	MRP-227, Rev. 1 Coverage Requirement
Core Support Barrel Assembly – Upper (core support Barrel) flange weld	C.5 Core Support Barrel Assembly Upper Flange Weld (UFW)	A minimum of 25% of the circumference of the UFW and adjacent base metal shall be examined
Core Support Barrel Assembly – Lower Cylinder Girth Welds	C6. Core Support Barrel Assembly – Middle Girth Weld (MGW)	A minimum of 25% of the OD circumference of the MGW and adjacent base metal shall be examined
Core Barrel Assembly – Upper core barrel flange weld	W3.Core Barrel Assembly – Upper flange weld (UFW)	A minimum of 25% of one side of the circumference of the surface of the UFW and adjacent base metal shall be examined
Core Support Barrel Assembly – Upper and Lower Core Barrel Cylinder Girth Welds	W4. Core Barrel Assembly – Lower Girth Weld (LGW)	A minimum of 25% of the OD circumference of the LGW and adjacent base metal shall be examined

For the original items in MRP-227-A, Note 4 clarified that a minimum of 75 percent of the total weld length (examined + unexamined) including coverage consistent with the Expansion criteria in [Table 5-2 or table 5-3], must be examined from either the inner or outer diameter for inspection credit. In MRP-227, Rev. 1, Note 5 to Table 4-2 and Note 8 to Table 4-3 state that "Examination coverage requires 25% of the circumference of either the inside diameter or the outside diameter of the weld." Note 6 to Table 4-2 and Note 10 to Table 4-3 state that "The stated coverage requirement is the minimum if no significant indications are found. However, the Examination Acceptance criteria in Section 5 require that additional coverage must be achieved in the same outage if significant flaws are found. This contingency should be considered for inspection planning purposes."

MRP-227, Rev. 1 contains a discussion of the inspection strategy for Westinghouse/Combustion Engineering (CE) core barrel weld sampling on p. 4-10 through 4-11 that pertains to the welds listed above. The discussion focuses on two elements: (1) A discussion of the probability of detecting an active cracking mechanism if a 25 percent sample of the weld is examined, on both a single plant and fleet-wide basis; (2) Focusing the 25 percent sample on the accessible portion of the weld most likely to exhibit cracking.

The NRC staff has several concerns related to the reduction of the required examination coverage for the welds listed in Table 1:

- The NRC staff is concerned that the reduced examination coverage is insufficient to provide reasonable assurance of component functionality considering that these welds are high consequence of failure items, which are not part of a redundant population.
- The discussion on pages 4-10 to 4-11 of MRP-227, Rev. 1 appears to describe some elements of a technical basis, but more detail is needed by the NRC staff to determine the adequacy of the technical basis.
- A determination of the most likely accessible portion of the weld to experience cracking is not required in Table 4-2 or Table 4-3. The discussion on pages 4-10 to 4-11 is not part of the report designated as NEI 03-08 "needed" guidance. Therefore, there is no guarantee licensees would perform such a determination.
- Even if a determination of the most likely accessible portion of the weld to experience cracking, is made, there may be significant uncertainty associated with such a determination and cracking may still be more likely to initiate in inaccessible portions, such as the weld ID.
- Coverage appears to be inconsistent between the UFW for the CE versus Westinghouse designs, with CE apparently required to examine both sides and Westinghouse only one side.

The NRC staff, therefore, requests the following information:

- Provide a technical justification for the reduction in the required examination coverage from 100 percent (minimum 75%) to 25 percent, for the component items listed in Table 1. If the technical justification relies in whole or part upon a statistical analysis, provide the detailed statistical analysis. The technical justification for the reduction in examination coverage should provide reasonable assurance that (1) the functionality of the core barrel will be maintained and (2) the structural integrity of the core barrel will be maintained to ensure safe shutdown of the reactor during the period of extended operation (PEO).
- Clarify whether the justification for reduction in the required examination coverage relies on the assumption that licensees will perform a plant-specific determination of the most likely portion of the weld to experience cracking.
- Discuss how it can be assured that the 25 percent sample of each weld examined will be selected based on an evaluation of the most likely accessible portion of the weld to exhibit cracking, since Table 4-2 and 4-3 do not require such an evaluation.
- Discuss how the proposed 25 percent sample examination coverage accounts for the possibility of cracking initiating on the opposite side of the weld from the side examined or in a portion of the component that is inaccessible.
- For C5., "Core Support barrel Assembly Upper Flange Weld (UFW)," clarify whether 25 percent of bolt sides of the weld are to be examined. If both sides are to be examined, explain the

inconsistency with W3. Core Barrel Assembly UFW, for which MRP-227, Rev. 1 only requires one side to be examined.

3.1.2 Industry Response

1. MRP-227, Revision 1 is based on a lead component and sampling approach to managing the aging-related degradation in the reactor vessel internals. The details of this approach are provided in Section 3.3 of MRP-227, Revision 1 [3]. The core barrel welds that are the subject of this question are primary components in the aging management strategy because they are considered to be lead components for their respective degradation mechanisms. These core barrel welds have not shown evidence of aging degradation to date in either the MRP-227-A inspections or the regular 10-year ISI VT-3 inspections that have been conducted. Per the definition of a primary item provided in Section 3.3.1, "where little or no service degradation has been experienced to date and/or service degradation is not expected solely based on the aging mechanism, a sample strategy for primary components is specified." This favorable operating experience was the first factor influencing the decision to reduce the required inspection coverage of these welds.

Statistical Basis for Reduced Core Barrel Weld Coverage

A sampling strategy for examination of the core barrel welds is dependent on several factors:

- Length of an acceptable crack
- Minimum size of a detectable crack
- Number of acceptable cracks permitted
- Expected distribution or location of cracking
- Inspection coverage on any given core barrel weld
- Number of welds inspected (at any given plant and across multiple plants)
- Effect of past inspection operating experience

A statistical evaluation of the required inspection coverage of 25% in MRP-227, Revision 1 was performed to support the discussion provided in Section 4.3. For simplicity, assumptions were made to reduce the number of factors that must be addressed.

- Length of an acceptable crack: The effect of this length was minimized in the analysis by assuming the presence of a very small crack and then assuming that this crack is at the detection limit of the visual inspection. A crack length of 0.25 inches was assumed in this case, expected to be within the detectable limits of an EVT-1 inspection, which has a required resolution demonstration using 0.044 inch characters per the MRP-228, Revision 2 Inspection Standard for Pressurized Water Reactor Internals [12]. Analyses for larger cracks were also performed and would provide more margin in this detectability analysis.

- Minimum size of a detectable crack: Covered in the "Length of an acceptable crack" bullet. Assuming a larger crack provides a plant with more margin in this detectability analysis, but can be more difficult to find acceptable by engineering evaluation.
- Number of acceptable cracks permitted: Multiple cases were evaluated to determine the sensitivity to this variable
- Expected distribution or location of cracking: Assumed that the potential cracks were randomly distributed around the circumference of any given weld (see response to part 2 of this question)
- Inspection coverage on any given core barrel weld: Assumed to be 25% of the entire weld length, per the requirements of MRP-227, Revision 1
- Number of welds inspected (at any given plant and across multiple plants): Multiple cases were evaluated to determine the sensitivity to this variable
- Effect of past inspection operating experience: Addressed in a conservative manner by treating those inspections as if they had only achieved the 25% coverage required in MRP-227, Revision 1.

The probability that at least one crack is detected was evaluated with these input parameters. This probability was calculated by simply evaluating the probability of detecting a single crack in a single weld and then extending it to multiple cracks in each weld or inspections across multiple welds. The effect of crack size was accounted for by recognizing that a crack will be detected if at least a part of the crack as long as the minimum detectable crack size intersects the inspected length. This results in some increase in detection probability with increasing crack size.

To extend the analysis to welds with more than one crack, it was assumed that the individual statistical trials were independent of one another and that each had the same probability of detection as the first crack. This assumption allows cracks to overlap one another, which is conservative relative to reality—a weld with 8 cracks present would have those cracks in 8 separate locations, resulting in a decrease of the length of weld available for each subsequent crack.

Similar assumptions were made for extending the analysis to multiple welds, whether in the same plant (e.g., if the upper flange weld and upper girth weld were inspected on the same core barrel) or across multiple plants (e.g., if upper flange welds were inspected at 10 different plants). Note that this assumes that certain welds within the same core barrel and across different plants are in the same statistical population. This was limited by the applicable degradation mechanisms: welds that are subject to SCC should be treated in a population separate from welds subject to IASCC. Thus, the core beltline welds were grouped together in one population and the welds outside of the core beltline were grouped in a separate population. It is assumed that each weld has a reasonably similar likelihood of forming a crack. Given that there has been no operating experience with SCC or IASCC cracking in the PWR core barrel welds, to date, making a different assumption about the likelihood is difficult and would be strongly based on speculation and hypothesis. Note that the discussion on stress, potential fabrication defects, and neutron dose provided in the other parts of the response to this question provide further support for this assumption about the population of welds.

The probability of detecting one crack in one inspected weld was calculated as noted above. The probability of interest for welds with multiple assumed cracks or across multiple inspected welds is the probability of detecting **at least one** of the cracks. This could be detecting one crack or 10 cracks or all of the cracks. Thus, it is the complement of the probability of detecting nothing and is calculated as:

$$1 - p_0^{\,n}$$

Where:

p_0 = probability of detecting nothing for a single crack or single inspection

n = number of cracks in a particular weld or number of welds inspected

The results of the probabilistic analysis are provided in Table 1, Table 2, and Table 3 for 0.25 inch, 1 inch, and 2 inch cracks, respectively. In these tables, the number of cracks across the columns indicates the number of cracks assumed in each weld inspected, while the number of inspections across the rows indicates the number of welds inspected. It is assumed that each weld inspected for the same mechanisms and to the same inspection standard (MRP-228 [12]) has the same likelihood of detecting a crack. Note that the probabilities calculated in these tables reduce the probability of detection due to the assumed minimum detectable length by subtracting that length from each end of the inspected length.

These tables show several things about the likelihood of crack detection by a 25% inspection coverage:

- Assumed crack size has a slight effect on the probability of detection, with larger cracks having a higher probability than smaller cracks. In this calculation, the increase in probability comes from the potential to detect the end of a crack that extends into the uninspected portion of the weld.
- Increased numbers of cracks in a given weld results in significant increases in the likelihood of a 25% inspection detecting at least one crack. As noted in Section 4.3 of MRP-227, Revision 1, the presence of eight cracks in a single weld results in at least a 90% probability of detecting at least one crack.
- Higher numbers of inspected welds has the same impact on the likelihood of detection.

Assuming one 0.25 inch flaw in each weld results in the likelihood of detection reaching 95% around 10 inspections with 25% coverage. To date, greater than 10 MRP-227 inspections have been performed on the welds subject to SCC and at least 10 inspections have been performed on welds subject to IASCC. At a high level, those inspections and the coverage achieved are summarized here:

- Westinghouse and CE upper flange weld: At least 11 inspections have been conducted, which have achieved 100% reported coverage of the weld. These have each been of one

side of the core barrel with a mix of inspections from the OD and inspections from the ID.

- Westinghouse lower flange weld and CE lower girth weld: At least 9 inspections have been conducted, which have achieved greater than 75% reported coverage of the weld. Six of these achieved greater than 90% and three reported 100% coverage. These have all been conducted from the OD of the core barrel.
- Westinghouse and CE upper girth weld: At least 9 inspections have been conducted, which have achieved 100% reported coverage of the weld. These have each been of one side of the core barrel with a mix of inspections from the OD and inspections from the ID.
- Westinghouse lower girth weld and CE middle girth weld (core beltline welds): At least 9 inspections have been conducted, which have achieved greater than 55% reported coverage of the weld. Seven of these achieved greater than 75% coverage. These have all been conducted from the OD of the barrel.

None of these inspections have reported cracking-related relevant conditions. In addition, ASME Code Section XI general visual (VT-3) inservice inspections (ISI) have been performed periodically at each plant at 10-year intervals during the initial 40-year licensing period. These inspections provided added assurance that no gross degradation is present in the core barrel weld surfaces that were examined.

These inspections provide a significant body of evidence for the lack of SCC or IASCC occurrence in the core barrel welds to date. This lack of an active degradation mechanism supports the use of a sampling strategy to monitor for the potential initiation of the degradation. These inspections achieved a much higher coverage level than the 25% assumed in Table 1 through Table 3, providing additional support for the statistical basis of the sampling strategy.

Table 1: Probability of Detecting at Least One Crack when Inspecting 25% of the Core Barrel Weld and Assuming 0.25 inch Cracks

Crack Length (in)	0.25						
No. of welds/ Inspections	1 crack	2 cracks	3 cracks	4 cracks	5 cracks	8 cracks	10 cracks
1	24.9%	43.7%	57.7%	68.3%	76.2%	89.9%	94.3%
2	43.7%	68.3%	82.1%	89.9%	94.3%	99.0%	99.7%
3	57.7%	82.1%	92.4%	96.8%	98.6%	99.9%	100.0%
4	68.3%	89.9%	96.8%	99.0%	99.7%	100.0%	100.0%
5	76.2%	94.3%	98.6%	99.7%	99.9%	100.0%	100.0%
6	82.1%	96.8%	99.4%	99.9%	100.0%	100.0%	100.0%
7	86.6%	98.2%	99.8%	100.0%	100.0%	100.0%	100.0%
8	89.9%	99.0%	99.9%	100.0%	100.0%	100.0%	100.0%
9	92.4%	99.4%	100.0%	100.0%	100.0%	100.0%	100.0%
10	94.3%	99.7%	100.0%	100.0%	100.0%	100.0%	100.0%
11	95.7%	99.8%	100.0%	100.0%	100.0%	100.0%	100.0%
12	96.8%	99.9%	100.0%	100.0%	100.0%	100.0%	100.0%
13	97.6%	99.9%	100.0%	100.0%	100.0%	100.0%	100.0%
14	98.2%	100.0%	100.0%	100.0%	100.0%	100.0%	100.0%
15	98.6%	100.0%	100.0%	100.0%	100.0%	100.0%	100.0%
16	99.0%	100.0%	100.0%	100.0%	100.0%	100.0%	100.0%
17	99.2%	100.0%	100.0%	100.0%	100.0%	100.0%	100.0%
18	99.4%	100.0%	100.0%	100.0%	100.0%	100.0%	100.0%
19	99.6%	100.0%	100.0%	100.0%	100.0%	100.0%	100.0%
20	99.7%	100.0%	100.0%	100.0%	100.0%	100.0%	100.0%

Table 2: Probability of Detecting at Least One Crack when Inspecting 25% of the Core Barrel Weld and Assuming 1 inch Cracks

Crack Length (in)	1						
No. of welds/ Inspections	1 crack	2 cracks	3 cracks	4 cracks	5 cracks	8 cracks	10 cracks
1	25.1%	43.9%	58.0%	68.6%	76.5%	90.1%	94.5%
2	43.9%	68.6%	82.4%	90.1%	94.5%	99.0%	99.7%
3	58.0%	82.4%	92.6%	96.9%	98.7%	99.9%	100.0%
4	68.6%	90.1%	96.9%	99.0%	99.7%	100.0%	100.0%
5	76.5%	94.5%	98.7%	99.7%	99.9%	100.0%	100.0%
6	82.4%	96.9%	99.5%	99.9%	100.0%	100.0%	100.0%
7	86.8%	98.3%	99.8%	100.0%	100.0%	100.0%	100.0%
8	90.1%	99.0%	99.9%	100.0%	100.0%	100.0%	100.0%
9	92.6%	99.5%	100.0%	100.0%	100.0%	100.0%	100.0%
10	94.5%	99.7%	100.0%	100.0%	100.0%	100.0%	100.0%
11	95.8%	99.8%	100.0%	100.0%	100.0%	100.0%	100.0%
12	96.9%	99.9%	100.0%	100.0%	100.0%	100.0%	100.0%
13	97.7%	99.9%	100.0%	100.0%	100.0%	100.0%	100.0%
14	98.3%	100.0%	100.0%	100.0%	100.0%	100.0%	100.0%
15	98.7%	100.0%	100.0%	100.0%	100.0%	100.0%	100.0%
16	99.0%	100.0%	100.0%	100.0%	100.0%	100.0%	100.0%
17	99.3%	100.0%	100.0%	100.0%	100.0%	100.0%	100.0%
18	99.5%	100.0%	100.0%	100.0%	100.0%	100.0%	100.0%
19	99.6%	100.0%	100.0%	100.0%	100.0%	100.0%	100.0%
20	99.7%	100.0%	100.0%	100.0%	100.0%	100.0%	100.0%

Table 3: Probability of Detecting at Least One Crack when Inspecting 25% of the Core Barrel Weld and Assuming 2 inch Cracks

Crack Length (in)	2					
No. of welds/ Inspections	1 crack	2 cracks	3 cracks	4 cracks	5 cracks	8 cracks	10 cracks
1	25.3%	44.3%	58.4%	68.9%	76.8%	90.4%	94.6%
2	44.3%	68.9%	82.7%	90.4%	94.6%	99.1%	99.7%
3	58.4%	82.7%	92.8%	97.0%	98.8%	99.9%	100.0%
4	68.9%	90.4%	97.0%	99.1%	99.7%	100.0%	100.0%
5	76.8%	94.6%	98.8%	99.7%	99.9%	100.0%	100.0%
6	82.7%	97.0%	99.5%	99.9%	100.0%	100.0%	100.0%
7	87.1%	98.3%	99.8%	100.0%	100.0%	100.0%	100.0%
8	90.4%	99.1%	99.9%	100.0%	100.0%	100.0%	100.0%
9	92.8%	99.5%	100.0%	100.0%	100.0%	100.0%	100.0%
10	94.6%	99.7%	100.0%	100.0%	100.0%	100.0%	100.0%
11	96.0%	99.8%	100.0%	100.0%	100.0%	100.0%	100.0%
12	97.0%	99.9%	100.0%	100.0%	100.0%	100.0%	100.0%
13	97.8%	99.9%	100.0%	100.0%	100.0%	100.0%	100.0%
14	98.3%	100.0%	100.0%	100.0%	100.0%	100.0%	100.0%
15	98.8%	100.0%	100.0%	100.0%	100.0%	100.0%	100.0%
16	99.1%	100.0%	100.0%	100.0%	100.0%	100.0%	100.0%
17	99.3%	100.0%	100.0%	100.0%	100.0%	100.0%	100.0%
18	99.5%	100.0%	100.0%	100.0%	100.0%	100.0%	100.0%
19	99.6%	100.0%	100.0%	100.0%	100.0%	100.0%	100.0%
20	99.7%	100.0%	100.0%	100.0%	100.0%	100.0%	100.0%

Functional Basis for Reduced Coverage Requirement

The preceding discussion in this response focused on the likelihood of flaw detection given the operating experience and the 25% coverage inspection specified by MRP-227, Revision 1. This included the details of the statistical argument presented in MRP-227, Revision 1 Section 4.3. These arguments are further supported by considering the potential for an effect on the function of the core barrel welds. According to WCAP-17096-NP-A [5], the function of the core barrel girth welds is to act as a primary core support structure. The pressurized water reactor issue management table in MRP-156 [7] provides additional detail on the core barrel functions:

- Provides core support through other internals structures attached to the barrel, such as the core lower core plate and baffle-former assembly
- Directs coolant flow to the core and out of the vessel
- Maintain the capability to insert the controls for safe shutdown through the lower core plate and providing alignment for the upper core plate

The discussion here will be divided into consideration of function during a **faulted** event and function during **normal** operation.

Faulted Event:

During a faulted event, such as an earthquake or loss of coolant accident, the stress applied to the core barrel is significantly higher than during normal operation. The reactor vessel internals are designed to maintain the functions of the core barrel even if the faulted condition results in complete 360°, through-wall fracture of a core barrel girth weld. The secondary core support structure in Westinghouse-designed plants and the core stops in CE-designed plants are designed to catch the core barrel and only allow a short drop of the core barrel if it fully fractures. The radial keys and clevises in Westinghouse-designed plants and core stabilizer lugs and snubbers in CE-designed plants at the bottom of the core barrel ensure that a barrel experiencing this type of extreme event cannot swing, rotate, or displace significantly. The combination of these two design features ensures that the core will be supported during a faulted event and that the control rods can still be inserted for safe shutdown.

Testing was conducted to measure the effect of various abnormal conditions on the ability to insert the control rods and the time to scram [16]. One of these tests investigated the effect of a full core drop type accident. As noted above, the distance that the core can drop is limited by the reactor vessel internals design. This limited drop distance leaves the upper fuel alignment pins partially engaged, which in turn limits the amount that the top nozzles of the fuel can be offset from the control rod clusters in the upper internals. Tests were performed at the maximum possible offset and determined that the control rods would still insert fully and that the increased scram times were within acceptable limits. These tests provide objective evidence of the ability to safely shut down during a faulted event where a core barrel girth weld completely separates with a 360°, through-wall crack.

A 360°, through-wall crack is not the only postulated failure. A through-wall crack that has propagated around most of the barrel but left a small remaining ligament was also considered. In this case, it was assumed that the remaining uncracked ligament was small enough to allow the rest of the barrel to separate and tilt, while still holding the side with the ligament in place. This case is considered less limiting than the full, 360°, through-wall crack for the following reasons:

- The existing design features that limit movement of the core barrel also limit the amount of possible tilt, in particular the radial keys and clevises or core stabilizing lugs and snubbers and the core barrel outlet nozzles

- The upper fuel alignment pins will still be engaged and prevent lateral movement beyond that already tested for the full core drop

- The presence of the remaining ligament of the core barrel will limit the amount of lateral movement that can occur and will ensure that the side of the core closest to the ligament remains well-aligned

The testing performed in [16] also tested the effect of significant fuel deflections (i.e., the center of the fuel assembly was deflected laterally while the top and bottom were pinned) and determined that effect on scram time was acceptable. This provides evidence that the small

"bend" in the control rod insertion path that could be caused by a tilted core barrel would not have an impact on the ability to insert the control rods for core shutdown.

The occurrence of a faulted event would result in plant shutdown and corrective actions to address the potential effects. Thus, continued operation in the degraded conditions considered here would not occur.

Normal Operation:

During normal operation, it is considered likely that a full separation of one of the girth welds would be detected through the loose parts monitoring system, in-core or ex-core detectors, or some other means. However, it is possible that the separation may not be recognized and addressed instantaneously since such a hypothetical event in this case would not be associated with the faulted conditions discussed above. From a safety standpoint, the separation would not result in a loss of core support or control rod insertability, but the effect of the separation on the bypass flow in the core region could have an impact on safety. Thus, the ability of the MRP-227, Revision 1 inspection to detect structurally significant cracking is important.

The primary stresses on the core barrel girth welds during normal operation are low. Based on existing analyses, the critical crack length for core barrel girth welds under normal operating conditions is at least several feet. This was calculated using a linear elastic fracture mechanics approach and evaluating the applied stress intensity factor at the crack tip versus the fracture toughness of the material, using the margin factor of 2.77 from WCAP-17096-NP-A. This critical flaw size is around 10% of the total core barrel weld length, which provides further credence to the one-side inspection of the weld since such a long crack is unlikely to form without penetrating the full thickness of the weld. Note that consideration of faulted conditions results in a shorter critical crack length due to the higher loads during an event; however, the safety impacts of a faulted event have already been dealt with separately above.

Summary:

Consideration of the faulted case and the normal operation case provide assurance that the core barrel will maintain its core support and safe shutdown functions under faulted conditions and that the MRP-227, Revision 1 inspection will have a reasonable probability of detecting a crack.

Conclusions

In conclusion, the reduced coverage requirement is justified by the following points:

- A sampling strategy can be employed per MRP-227, Revision 1, Section 3.3.1 since "little or no service degradation has been experienced to date and/or service degradation is not expected solely based on the aging mechanism"
- The small crack size assumed for this evaluation is consistent with the capabilities of the inspection method deemed appropriate for this component (EVT-1)

- The crack size assumed for this evaluation is small enough to reasonably assure the continued functionality and structural integrity of the core barrel. A plant-specific acceptance criteria evaluation could be used to increase the allowable crack length or number of cracks to provide even more margin for this evaluation.
- Under faulted conditions, the design features included in the reactor vessel internals limit the adverse effects of a full failure of a core barrel girth weld on the core support or safe shutdown functions of the core barrel
- Under normal operating conditions, the critical crack size based on having a margin between the applied stress intensity and the fracture toughness of the material allows for a critical crack length of at least several feet, which increases the probability of detecting a structurally significant crack
- Multiple plants have performed inspections on these PWR core barrel welds and other PWR core barrel welds at higher coverage levels and have not detected cracking-related relevant conditions
- These PWR core barrel inspection requirements will continue to be applicable to plants licensed for the period of extended operation; thus, the sampling of PWR core barrel welds will continue to grow
- Per the requirements of MRP-227, Revision 1, Section 7, any relevant conditions detected during the inspections will be recorded and entered into the owner's plant corrective action program and dispositioned. Furthermore, the results of the inspection will be reported to the MRP Program Manager for publication in the industry report to the fleet, the regulator, the PWROG, and other stakeholders. Thus, if evidence of core barrel weld cracking is detected at one plant, it will result in further evaluation and response across the fleet.

2. The justification provided in the response to part 1 of this question was based on a random distribution of SCC or IASCC cracking in each weld. The likelihood of SCC or IASCC at any given location is expected to vary somewhat based on material, environment, or stress conditions.

   Quality assurance of these safety-related components during fabrication and construction would have reduced or eliminated the possibility of many potential material issues that can contribute to SCC or IASCC. It can be reasonably assumed that the base and weld metal composition and quality were consistent from weld to weld and plant to plant due to procurement and testing requirements. The non-destructive testing required for each weld during manufacturing, such as radiographic testing and dye penetrant testing of weld surfaces, would have detected significant weld defects that could have served as crack initiation sites. A review of early plant and late plant drawings showed that the non-destructive test requirements were similar. These fabrication and testing requirements would have been applicable to all core barrel manufacturers. It is possible that weld stops and starts, embedded slag or porosity, or locations of weld repairs exist, but the location of these potential flaw initiation sites is expected to be randomly distributed, consistent with the random distribution of cracking assumed in the response to part 1 of this question.

   Local variations in environmental conditions are not expected for water chemistry but are expected for accumulated neutron dose. The PWR core barrel welds are located in moderate to high flow areas with few opportunities for crevices. This flow refreshes the coolant in contact with the welds, avoiding the risk of SCC or IASCC occurring due to deleterious chemical species. Additionally, the primary water coolant chemistry is maintained according to the action levels in the EPRI Primary Water Chemistry Guidelines [9]. As noted in MRP-227, Revision 1, the neutron irradiation dose accumulated on the core barrel welds is going to vary depending on location. If IASCC cracking occurs, it might be expected to appear first in a location with the highest fluence and stress, with stress as the more important factor at higher fluence levels. However, the location of highest stress (barrel outer diameter due to thermal effects) does not correspond with the location of highest dose (barrel inner diameter due to core proximity) on the core barrel. Due to the lack of occurrences of IASCC cracking in PWR welded components to date it cannot be conclusively determined what combination of stress or dose will create the highest likelihood of IASCC. However, as noted in the MRP-227, Revision 1 discussion of the core barrel welds affected by IASCC (the Westinghouse-design LGW and CE-design MGW), only the outer diameter is accessible for inspection. Thus, the 25% sampling inspection of PWR core barrel welds for IASCC should include portions of the weld with the highest neutron fluence, within the accessibility limitations imposed by the structure. The location of more highly irradiated portions of the core barrel welds can be determined by locating the areas where the edge of the reactor core is radially closest to the barrel.

   Based on these considerations, the only requirement for focusing the inspection on areas with a higher likelihood of degradation is the requirement to include a sample of the most highly irradiated accessible portions of the CE middle girth weld and Westinghouse lower girth weld in the 25% of the weld inspected. Also, since the cracking is expected to be randomly distributed,

the inspection area chosen for these primary component welds with a 25% coverage requirement should be different for each inspection interval, ensuring more complete coverage of the welds over the sampling period. These two changes are reflected in the response to part 3 of this question.

3. For the CE middle girth weld and the Westinghouse lower girth weld, Table 4-2 and Table 4-3 will be updated to include the following text in the Examination Coverage column:

    - Westinghouse UFW, W3:
        - Note 11 will be added to the Examination Coverage of the table entry: "11. Inspections subsequent to the first 25% sampling inspection must cover weld length that was not inspected during previous examinations. A minimum of 20% out of the 25% inspection coverage of each subsequent inspection must include previously un-inspected weld length as possible within the limitation of the remaining accessible un-inspected weld."
    - CE UFW, C5:
        - Note 7 will be added to the Examination Coverage of the table entry: "7. Inspections subsequent to the first 25% sampling inspection must cover weld length that was not inspected during previous examinations. A minimum of 20% out of the 25% inspection coverage of each subsequent inspection must include previously un-inspected weld length as possible within the limitation of the remaining accessible un-inspected weld."
    - Westinghouse LGW, W4:
        - Examination Coverage text will be updated: "A minimum of 25% of the OD circumference of the LGW and adjacent base metal shall be examined. This 25% sample must include the accessible portion of the weld OD with the highest accumulated neutron fluence."
        - Note 11 will be added to the Examination Coverage of the table entry: "11. Inspections subsequent to the first 25% sampling inspection must cover weld length that was not inspected during previous examinations. A minimum of 20% out of the 25% inspection coverage of each subsequent inspection must include previously un-inspected weld length as possible within the limitation of the remaining accessible un-inspected weld."
    - CE MGW, C6:
        - Examination Coverage text will be updated: "A minimum of 25% of the OD circumference of the MGW and adjacent base metal shall be examined. This 25% sample must include the accessible portion of the weld OD with the highest accumulated neutron fluence."
        - Note 7 will be added to the Examination Coverage of the table entry: "7. Inspections subsequent to the first 25% sampling inspection must cover weld length that was not inspected during previous examinations. A minimum of 20% out of the 25% inspection coverage of each subsequent inspection must include previously un-inspected weld length as possible within the limitation of the remaining accessible un-inspected weld."

Note that in Westinghouse-designed plants with neutron panels on the outside of the core barrel, only a portion of the weld circumference in the core beltline region is accessible given currently available inspection techniques. The exact percentage of the core barrel circumference covered by the neutron panels varies with plant design. Between 50 and 60 percent of the barrel circumference will be accessible to a visual inspection in a neutron panel plant. This does not impact the inspection coverage requirements detailed here and in MRP-227, Revision 1 because greater than 25% of the total core barrel girth weld length is accessible.

4. Consideration of which side SCC or IASCC is expected to initiate from is different for the upper flange weld and the lower girth weld (Westinghouse) or middle girth weld (CE). This is due to the effects of neutron radiation.

   For the upper flange weld, the normal operating stresses are expected to generally be low, leaving the potential for SCC initiation driven primarily by the weld residual stresses. As discussed in the response to part 2 of this question, there may be variations in weld residual stress due to stops and starts or repairs, but these would be distributed randomly from plant to plant, both around the barrel circumference and between the OD and ID of the barrel. Additionally, since these were full penetration welds, the order of the welds made from the OD and those made from the ID could also have an impact on whether the residual stress is higher on the inside or the outside. This was addressed in the fabrication specifications by requiring that the welds sequences be scheduled such that barrel distortion and residual stresses be minimized. To meet this requirement, the manufacturers would have welded partway on one side and then switched to the other side to weld partway, alternating the imposition of stress due to weld shrinkage during cooling and solidification. The exact number of alternations may have varied from manufacturer to manufacturer, but the end goal of a minimizing distortion and weld residual stress would have driven each manufacturer to a similar end result where the weld residual stresses are similar on the ID and the OD. Based on these fabrication considerations, the likelihood of crack initiation is considered similar between the OD and the ID of the UFW and the current guidance allowing inspection of one side is appropriate.

   For the Westinghouse LGW or the CE MGW, the normal operating stresses are strongly driven by the thermal differential across the barrel wall due to the effects of irradiation. The higher temperature closer to the core causes a net tensile stress on the OD of the barrel in the region of the LGW or MGW that is higher than that on the ID (even if weld residual stresses are not assumed to relax). As discussed in the response to part 2 of this question, current data and experience does not conclusively demonstrate that the higher stress on the OD of the barrel will lead to more IASCC risk than the higher dose on the ID of the barrel. However, the ID of the barrel in this region would only be accessible if the baffle-former assembly (Westinghouse designs) or core shroud assembly (CE designs) were disassembled to allow access. This would be a high-risk, high-dose, high-cost alternative and would be unreasonable and unwarranted considering the complete lack of observed cracking in the operating experience with no concomitant increase to plant safety.

The one-sided inspection is consistent with the intention of the MRP-227, Revision 1 inspection and evaluation program to manage the potential degradation of the core barrel welds due to aging-related effects. Given the considerations presented in this response and in the response to part 2 of this question, the locations required in the inspection are reasonable for monitoring for cracking. Combining this with the response to part 1 of this question, the inspection sampling of multiple core barrel welds within a plant and across multiple plants provides a reasonable basis for expecting that the core barrel weld inspections will detect the appearance of cracking degradation in those welds. Once cracking is detected, the industry will respond with further measures that may include expanded inspections, volumetric inspections, additional analyses, or repair and mitigation operations.

5. The intention of the Primary component entry for the CE component C5: "Core Support Barrel Assembly Upper Flange Weld (UFW)" was to require inspection of one side of the weld. This could be conducted on either side of the weld (ID or OD). This would be consistent with the coverage requirements for the Westinghouse component W3: "Core Barrel Assembly Upper Flange Weld". The examination coverage for C5 will be updated to state:

   "A minimum of 25% of one side of the circumference of the surface of the UFW and adjacent base metal shall be examined."

3.2 Response to NRC RAI 9

3.2.1 NRC Question

In Table 4-2, CE Plants Primary Components, Item C8, "Lower Support Structure – Core Support Columns," is a new item that includes both core support columns (for plants with full height bolted core shroud plates) and core support column welds (for plants with half-height welded core shroud plates). The examination coverage for the core support columns is 25% of the column assemblies as visible using a VT-3 examination from above the lower core plate and for the core support column welds is 100 percent of the accessible surfaces. In MRP-227-A the equivalent item included only the core support column welds, with examination coverage of 100 percent of the accessible surfaces, for all plants. There are differences in required examination coverage for the core support column components for the two plant design variations. In addition, the component in Westinghouse-design RVI with the same function is an Expansion component whereas the CE core support columns are a Primary component.

MRP-227-A has two separate items for Westinghouse Lower Support Assembly - lower support column bodies depending on the material (cast or non-cast). In MRP-227, Rev. 1, these two items are combined into one in Item W4.4., "Lower Support Assembly – Lower Support Column Bodies (both cast and non-cast)." In addition, the examination method is changed from enhanced visual testing (EVT-1) examination to visual testing (VT)-3 examination and the examination coverage is changed from 100 percent of accessible surfaces (for non-cast) or 100 percent of accessible support columns (for cast) to 25 percent of column assemblies as visible using from above the lower core plate.

The NRC staff is concerned that the reduced coverage for the CE core support columns and Westinghouse lower support column bodies is not sufficient to provide reasonable assurance of component functionality, considering that the lower support columns are high consequence of failure components. Also, it is not clear how much information can be gained by a visual inspection from above the core plate.

To resolve these discrepancies, the NRC staff requests the following information:

a. Justify the required coverage of 25 percent as visible from above the core plate for Item C8 and W4.4 is sufficient to provide reasonable assurance of functionality.
b. Justify the use of VT-3 examination instead of EVT-1 to detect cracking.
c. Clarify the meaning of "25% of column assemblies as visible using a VT-3 examination from above the lower core plate." Does this mean that 1) only 25 percent of the total number of columns visible need to be inspected, 2) 25 percent of the total number of columns (visible and not visible) must be examined to claim credit for the examination, or that 3) 25 percent of the total columns should be inspected if this number is visible? Should all columns visible from above the core plate be examined, or just enough to constitute 25 percent of the total population (visible plus not visible).
d. What expansion of the examination scope to the remaining columns will be conducted if degradation is observed in the 25 percent sample?

e. For CE-design RVI, explain why examination of the core support columns is specified only for plants with full-height bolted shroud plates and not for plants with core shrouds assembled in two vertical sections.
f. Explain why the core support columns are a Primary component for CE plants but the component in Westinghouse plants with the same function (lower core support columns) is an Expansion component.

3.2.2 Industry Response

a. MRP-227, Revision 1 defined the examination coverage of the expansion component lower support column bodies as 25% of the column assemblies as visible using a VT-3 examination from above the lower core plate. The basis to reduce the examination coverage is the combination of the low likelihood of failure of the lower support columns with the significant redundancy provided in the lower support structure. PWROG-14048, Revision 2 [8] (Revision 1 provided to the NRC for information only via [10] with a staff assessment documented in [15]) was generated to develop a justification that the lower support columns will remain functional through the licensing renewal period of extended operation. To do this an FMEA was performed for the components followed by a failure tolerance analysis (considering both the Westinghouse and CE designs). The most limiting functionality case was determined by analysis to be a degraded condition where over 50 percent of the support columns were failed (for each of the column configurations based on plant design).

   Regarding the likelihood of failure, full section cracking of a column such that compressive load bearing capability is lost is considered to be an extremely unlikely scenario for the following reasons:

   - Quality controls during fabrication, such as liquid penetrant inspection or radiography, limit the number and size of the flaws that could be present.
   - Cast austenitic stainless steel lower support columns have been shown to have relatively low ferrite content such that susceptibility to thermal embrittlement and the combined effects of thermal and irradiation embrittlement are not expected to be dominant [13], which was endorsed by the NRC staff via [14].
   - The tensile stress in the columns is generally low (less than 15 ksi) under all conditions as demonstrated in PWROG-14048 [10]. Furthermore, this tensile stress is primarily a result of bending caused by displacement-limited loads, which is not conducive to cracking through the full section.
   - There are no credible mechanisms for flaw initiation or growth beyond what is permitted by the fabrication controls. A representative flaw tolerance evaluation shows critical flaw sizes to be more than ten times larger than any flaw permitted by fabrication.
   - Even for compressive loads transmitted across a complete tensile crack normal to the column axis, the material has sufficient remaining toughness to withstand any cracking from (transverse) tensile stresses that could be generated in the columns as a result of the tensile loading

The conclusion from the functionality analysis is that there is significant redundancy in the system such that greater than 50% of the columns can be non-load bearing and the core will remain adequately supported to allow the rods to be safely inserted.

Due to the combination of the low likelihood of failure and the significant redundancy in the column demonstrated through analysis (functionality is retained even after postulated loss of support from 50% of the columns), it is considered technically acceptable to sample the columns for evidence of aging-related degradation by performing an examination of 25% of the column bodies. This is consistent with the sampling strategy outlined in MRP-227, Revision 1 for components where little or no service degradation has been experienced to date and/or service degradation is not expected solely based on the aging mechanism. If no degradation is found within the 25% of the columns sampled, it can be reasonably concluded that there will be very few partially cracked columns in the uninspected 75% of the columns and that, given the substantial structural margins documented in PWROG-14048, the core support structure can be expected to retain adequate margin in its structural integrity. The examination would expand to the remainder of the column bodies should degradation be observed in the initial inspection population.

The inspection of the columns from above the core plate is justified based on the level of degradation required to cause loss of functionality. Per PWROG-14048 loss of column function can only be compromised by the loss of compressive load carrying capability that would only be result from full section cracking. Such loss of function would not be caused by small, tight cracks which could only be detected by a high resolution visual examination. Instead, this would result from the relevant conditions called out in MRP-227, Revision 1, Table 5-3: "fractured, misaligned, or missing columns." These relevant indications can be detected by a visual VT-3 examination from above the core plate.

b. As noted in the response to part A, the relevant conditions for the visual inspection are consistent with the level of degradation required to impact the functionality of the columns. A VT-3 inspection through the holes in the core plate is adequate to detect fully fractured, misaligned, or missing columns, which are the only condition that would impact functionality. The justification for the level of degradation necessary to affect functionality is provided in PWROG-14048. Additionally, the portions of the columns visible through the holes of the core plate are in regions of high fluence.

c. The intent of the examination coverage for the columns is consistent with option 2 in the question. The inspection requirement "25% of the column assemblies as visible using a VT-3 examination from above the lower core plate," refers to 25% of the total number of columns from the overall population (both those visible and those not visible when viewing from above the lower core plate) in order to claim credit for the inspection. This will be clarified by modifying the text in the MRP-227, Revision 1 tables. For the CE core support columns, the language in Table 4-2 of MRP-227, Revision 1 will be updated as shown after the response to part F of this RAI response (Note that the CE core support columns will be moved to the Expansion category

and placed in Table 4-5, as explained in part F). For the Westinghouse lower support columns, the text in Table 4-6 of MRP-227, Revision 1 will be updated as follows.

Table 4-6 of MRP-227, Revision 1 will be updated as shown below:

Expansion Item	Applicability	Effect (Mechanism)	Primary Link (Note 1)	Examination Method/Frequency (Note 1)	Examination Coverage
Lower Support Assembly W4.4.Lower support column bodies (both cast and non-cast)	All plants	Cracking (IASCC) Aging Management (IE)	W4.Lower Girth Weld (LGW)	Visual (VT-3) examination. Re-inspection every 10 years following initial inspection.	25% of the total number of column assemblies (both visible and non-visible from above the lower core plate) ~~as visible~~ using a VT-3 examination from above the lower core plate. See Figure 4-23.

d. As stated in the response to part a, should degradation be observed in the initial inspection population the examination would expand to include the remainder of the population of the column bodies (that are visible through the lower core plate). Table 4-5 of MRP-227, Revision 1 will be updated to make this clarification as shown below following the response to part F of this RAI.

e. Table 4-2 of MRP-227, Revision 1 lists the inspection of the core support columns as a Primary component inspection of the CE RVI for plants with either full-height or half-height welded core shroud designs. MRP-227, Revision 1 does specify 25% examination coverage of the core support columns for plants with a full height bolted core shroud design and 100% examination coverage of the accessible column assemblies for plants with core shrouds assembled in two vertical sections. However, the conclusions in PWROG-14048 provide sufficient technical justification for the core support column welds to be an Expansion component for both core shroud designs as discussed in the response to part A of this RAI. The conclusions of PWROG-14048 that inspections are not necessary to ensure the functionality of the lower support columns through the period of extended operation are applicable to both CE RVI core shroud designs. MRP-227, Revision 1 will be revised to change this component from Primary to Expansion as shown in the response to part F of this question below.

f. PWROG-14048 provides sufficient technical justification for the lower (core) support columns to be expansion components for both the Westinghouse and CE RVI design, based on the low likelihood of failure and the significant margin for functionality as described in the response to part A of this question. However, MRP-227, Revision 1 was issued for safety evaluation prior to the publication of PWROG-14048 in February 2017, so changes to make the CE core support columns into an expansion item were not implemented at that point. Assignment of the CE columns to the primary inspection category was carried over from MRP-227-A. Based on the justification provided in PWROG-14048, the CE core support columns will be changed to Expansion components from the core barrel middle girth weld (MGW) similar to the expansion of the Westinghouse lower support column bodies from the core barrel lower girth weld (LGW). The changes to be implemented are shown in the following tables.

The core support columns (Item C8) will be moved from Table 4-2 (CE Plant Primary Components) to Table 4-5 (CE Plant Expansion Components). Changes from the original content of this item from Table 4-2 are shown below.

Expansion Item	Applicability	Effect (Mechanism)	Primary Link (Note 1)	Examination Method/Frequency (Note 1)	Examination Coverage
C8.Lower Support Structure Core support columns	Plants with full-height bolted or half-height welded core shroud plates	Cracking (SCC, IASCC, Fatigue including damaged or fractured material) Aging Management (IE, TE)	C6.Middle Girth Weld (MGW)	Visual (VT-3) examination Re-inspection every 10 years following initial inspection.	Plants with full height bolted core shroud plates: 25% of the total number of column assemblies (both visible and non-visible from above the core support plate) ~~as visible~~ using a VT-3 examination from above the core support plate. Plants with core shrouds assembled in two vertical sections: 25% ~~100%~~ of the accessible surfaces of the core support column welds, from the top side of the core support plate ~~(Note 3)~~. (Note 3) See Figures 4-36 and 4-374-~~31~~.

3. The stated minimum coverage requirement is the minimum if no significant indications are found. However, the Examination Acceptance criteria in Section 5 require that additional coverage must be achieved in the same outage if significant flaws are found. This contingency should be considered for inspection planning purposes.

Table 5-2 of MRP-227, Revision 1 is then updated as shown below to reflect that the core support column inspection is now an Expansion component inspection and will be expanded to 100% should an indication be found in the 25% population.

Primary Item	Applicability	Examination Acceptance Criteria (Note 1)	Expansion Link(s)	Expansion Criteria	Expansion Item Examination Acceptance Criteria
Core Support Barrel Assembly Middle Girth Weld (MGW)	All plants	Visual (EVT-1) examination. The specific relevant condition is a detectable crack-like surface indication.	Middle Axial Weld (MAW) Lower Axial Weld (LAW) Core Support Columns	The confirmed detection of a surface-breaking linear indication in the MGW shall require that the inspection coverage of the MGW be extended to include 100% of the accessible length during the same outage. The confirmed detection and sizing of a surface-breaking indication with a length greater than two inches in the MGW shall require that the inspection be expanded to include the MAW and LAW by the completion of the next refueling outage. The confirmed detection of a surface-breaking linear indication in the MGW shall require examination of 25% (of the total of both visible and non-visible as seen from above the core support plate) of the core support columns assemblies by the completion of the next refueling outage.	The specific relevant condition for the expansion lower cylinder axial welds is a detectable crack-like surface indication. The specific relevant condition for the core support columns welds is a disruption or discontinuity in the surface of the weld. The specific relevant condition for the core support columns viewed from above the core support plate is missing or separated welds, or fractured, misaligned or missing columns.

Table 4-6 of MRP-227, Revision 1 is then updated as shown below to note that the lower support column bodies Expansion inspection will be expanded to 100% should an indication be found in the 25% population by addition of Note 3 to the table.

Lower Support Assembly W4.4.Lower support column bodies (both cast and non-cast)	All plants	Cracking (IASCC) Aging Management (IE)	W4.Lower Girth Weld (LGW)	Visual (VT-3) examination. Re-inspection every 10 years following initial inspection.	25% of the total number of column assemblies (both visible and non-visible from above the lower core plate) using a VT-3 examination from above the lower core plate. (Note 3) See Figure 4-23.

3. The stated minimum coverage requirement is the minimum if no significant indications are found. However, the Examination Acceptance criteria in Section 5 require that additional coverage must be achieved in the same outage if significant flaws are found. This contingency should be considered for inspection planning purposes.

3.3 Response to NRC RAI 10

3.3.1 NRC Question

In Table 4-2, for Item C12., "Lower Support Structure – Deep Beams," and Table 4-5,

Item C5.4., "Lower Support Structure - Lower Core support Beams," the examination coverage has been changed to 25 percent of the total number of beam-to-beam welds. The examination coverage in MRP-227-A for the Lower Support Structure - Deep Beams does not specify a percentage of beam-to-beam welds that must be examined, but it is implied that 100 percent of the welds should be examined. The examination coverage in MRP-227-A for the lower core support beams is 100 percent of accessible surfaces. Because both components are high consequence of failure components, the NRC staff is concerned that the reduced examination coverage is insufficient to ensure functionality of the components.

The NRC staff therefore requests the following information:

a. Provide a justification for the reduction in coverage for these two items. The technical justification for the reduction in examination coverage should provide reasonable assurance that (1) the functionality of the components will be maintained and (2) the structural integrity of the components will be maintained to ensure safe shutdown of the reactor during the PEO.

b. What expansion to the remaining beam-to-beam welds will be conducted if degradation is found in the initial 25 percent inspection sample?

3.3.2 Industry Response

a. Item C12: Lower Support Structure – Deep Beams:

   Per MRP-227, Revision 1, the deep beams are only applicable to the CE plants with welded core shrouds assembled from full-height shroud plates. The deep beams are located directly beneath the core and the fuel assemblies sit on top of the beams. Figure 1 shows an approximate sketch of the deep beams (the same figure was included in MRP-227-A). The circles labeled "1" and "2" are the fuel alignment pins that interface with the fuel assemblies. The arrow labeled "3" points to one of the joints between the beams that make up the assembly. A full-penetration weld is located at each of these joint locations. These are the welds that are included in the Primary component inspection requirement of CE Item C12. The previous coverage requirement under MRP-227-A included all of the beam welds and required EVT-1 inspection of the top 4 inches of each weld. The coverage requirement under MRP-227, Revision 1 is to examine 25% of the total number of beam-to-beam welds, with the coverage including the top 4 inches of each weld. There is one full-penetration weld at each beam –to-beam intersection in the structure. Between 150 and 200 of these intersections/welds are expected to be accessible for the inspection, so the 25% coverage will include approximately 35 to 50 welds. At each location, this inspection

includes both sides of the weld, since they are full penetration welds and it is unknown which side would be more susceptible to degradation.

From the standpoint of likelihood of degradation, the applicable degradation mechanisms for these welds are fatigue and irradiation embrittlement (IE). The dose on the welds is highest at the top, hence the requirement to inspect the top 4 inches, and also higher toward the core centerline and lowest at the outer edges, radially. Similar to what was found for core support columns in the work supporting PWROG-14048-NP [8], the stress is expected to be highest near the outer edges, radially, of the assembly. This is due to the effects of thermal expansion. Because of these stress and radiation distributions, it is not clear where the most likely place for degradation to initiate is located. Thus, it is considered best to require the inspected locations be spread evenly across the deep beam structure to sample a variety of stress-dose combinations. Further, the welds included in the inspected sample should be different at each inspection interval, as possible within the constraints of accessibility and past inspection coverage.

The function of the deep beams is to directly support the core, to keep the fuel in place and to maintain alignment for control element assembly insertion. From the standpoint of functionality, the welded array is a redundant structure. If one weld of a cross-beam fails completely, the other end of that particular beam would still be attached to another main beam. The main beams are welded at multiple locations and would require multiple weld failures to compromise function. Assurance of the continued functionality of the deep beams is also aided by the fact that the onset of the loss of structural functionality would be likely to be first detected during fuel loading or unloading conducted during each refueling outage. The fuel loading and unloading operations are expected to detect this loss of functionality as misaligned fuel assemblies or abnormal difficulty with removing or placing fuel assemblies.

Given the number of individual welds that will be inspected for potential degradation, the redundancy in the deep beam structure, the monitoring for loss of function through fuel loading and unloading, and the applicability of the sampling approach described in MRP-227, reducing the inspections to 25% of the welds will provide adequate coverage to detect the onset of failures of significance. The reduced inspection coverage requirement of MRP-227, Revision 1 is justified for the Item C12: Deep Beams. Note that the coverage requirement will be modified to require that the inspection be spread out across the structure and be performed on different sets of welds after each inspection interval. The text for the “Examination Coverage” column of Table 4-2 in MRP-227, Revision 1 will be revised as follows:

> Examine 25% of the total number of beam-to-beam welds. Coverage on each weld examined includes the top four inches of the weld in the vertical orientation. The inspection coverage must be evenly distributed across the deep beam structure.

- o Note 7 will be added to the Examination Coverage of the table entry: “7. Inspections subsequent to the first 25% sampling inspection must cover weld length that was not inspected during previous examinations. A minimum of 20% out of the 25% inspection coverage of each subsequent inspection must include previously un-

inspected weld length as possible within the limitation of the remaining accessible un-inspected weld."

- Note 8 will also be added to the Examination Coverage of the table entry: "8. Justification that adequate distribution of the inspection coverage has been achieved can be based on geometric or layout arguments. For example, since each rectangular axial compartment in the structure has four welds around it, which are shared with the adjacent compartments on two sides, the inspection coverage could consist of the selection of one weld from every second compartment in the structure."

Item C5.4, "Lower Support Structure - Lower Core support Beams"

Item C5.4 is applicable to all CE plants except those with welded core shrouds assembly with full-height core shroud plates. These beams are located at the bottom of the core barrel directly beneath the core support columns (see Figure 4-35 in MRP-227, Revision 1). The beams function to support the core support columns which in turn support the core support plate which supports the fuel.

The lower core support beams are redundant components similar to the core support columns. The evaluation documented in PWROG-14048-NP showed that greater than 50 percent of the core support columns could be degraded without a loss of function. Since the lower core support beams provide the structure to support the core support columns, it is expected that a similar level of degradation could be tolerated in the beams. This margin to loss of function provides the technical justification for the reduced inspection coverage requirement of MRP-227, Revision 1.

It must also be noted that the location and geometry of the lower core support beams presents significant challenges to achieving higher coverage levels. Increased coverage levels such as 75% or 100% are likely not attainable.

b. If degradation is found in the initial 25 percent inspection population, the examination would be expanded to include the remaining beam-to-beam welds. The deep beam line item in Table 4-2 of MRP-227, Revision 1 will be updated to include the reference to the existing Note 6 in the table, which specifies that the stated coverage requirement is the minimum if no significant indications are found.

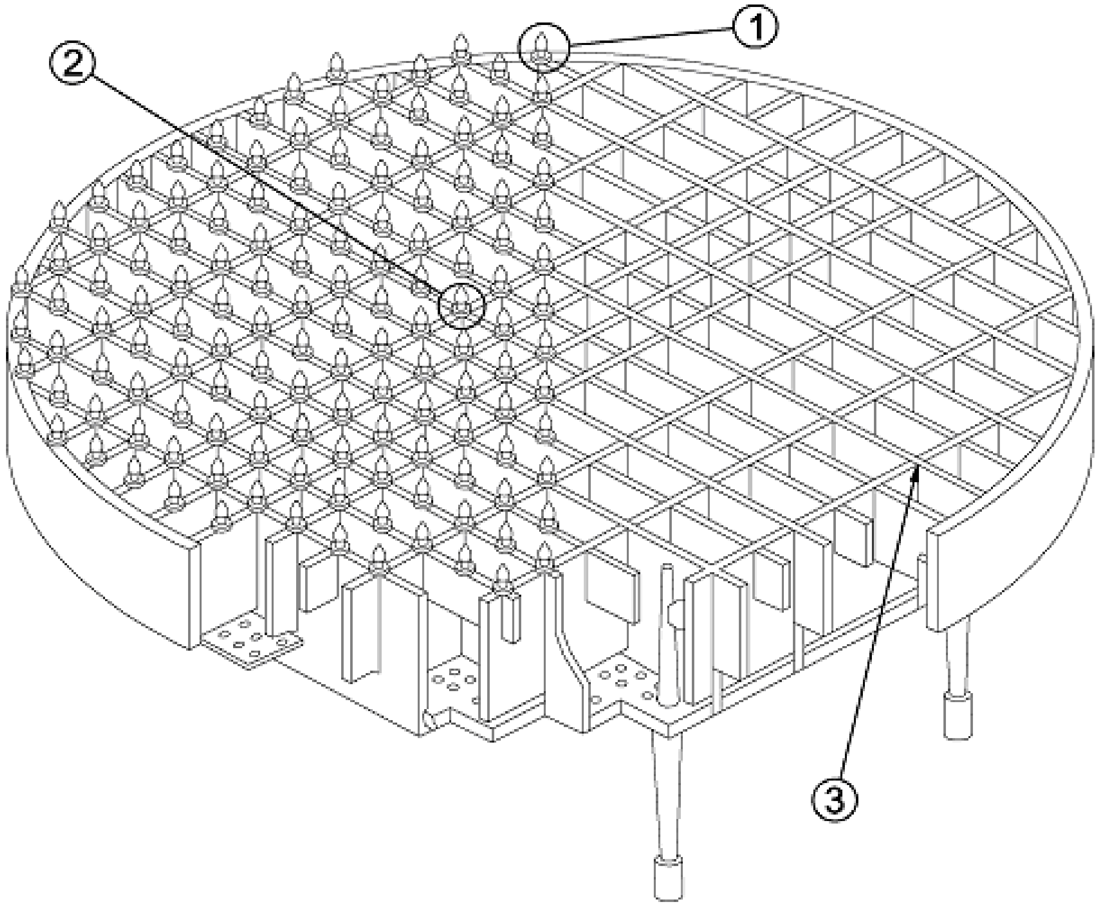

Figure 1: Approximate schematic sketch of the deep beam structure (figure from MRP-227-A)

Table 4-2 of MRP-227, Revision 1 will be updated as shown below to reflect this change.

Primary Item	Applicability	Effect (Mechanism)	Expansion Link (Note 1)	Examination Method/Frequency (Note 1)	Examination Coverage
C12.Lower Support Structure Deep beams	Only plants with welded core shrouds assembled with full-height shroud plates	Cracking (Fatigue) that results in a detectable surface-breaking indication in the welds or beams. Aging Management (IE)	None	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 adequacy of remaining fatigue life cannot be demonstrated.	Examine 25% of the total number of beam-to-beam welds. Coverage on each weld examined includes the top four inches of the weld in the vertical orientation. The inspection coverage must be evenly distributed across the deep beam structure. (Notes 6, 7 and 8) See Figure ~~4-34~~4-33.

7. Inspections subsequent to the first 25% sampling inspection must cover weld length that was not inspected during previous examinations. A minimum of 20% out of the 25% inspection coverage of each subsequent inspection must include previously un-inspected weld length as possible within the limitation of the remaining accessible un-inspected weld.
8. Justification that adequate distribution of the inspection coverage has been achieved can be based on geometric or layout arguments. For example, since each rectangular axial compartment in the structure has four welds around it, which are shared with the adjacent compartments on two sides, the inspection coverage could consist of the selection of one weld from every second compartment in the structure.

Table 5-2 of MRP-227, Revision 1 is updated as follows to reflect the expansion of the deep beams to 100% of the component population if degradation is observed in the initial exam.

Primary Item	Applicability	Examination Acceptance Criteria (Note 1)	Expansion Link(s)	Expansion Criteria	Expansion Item Examination Acceptance Criteria
Lower Support Structure Deep beams	Only plants with welded core shrouds assembled with full-height shroud plates	Visual (EVT-1) examination. The specific relevant condition is a detectable crack-like indication.	~~None~~ Remaining deep beams	~~N/A~~ Confirmed evidence of a detectable crack-like indication in one or more of the deep beams examined in the 25% population selected for the Primary component inspection shall require the inspection coverage to be expanded to the remaining deep beams by the completion of the next refueling outage.	~~N/A~~ The specific relevant condition is a detectable crack-like indication.

3.4 Response to NRC RAI 12

3.4.1 NRC Question

In Table 4-5, for plant designs with core shrouds assembled with full-height shroud plates, the core shroud assembly, remaining axial welds, ribs and rings has been split into two items: C3.1, “Remaining axial welds,” and C3.2, “Ribs and rings.” The coverage for these two items is different, 75 percent for the remaining axial weld length and adjacent base metal as visible from the core side of the shroud other than that already inspected under the primary link, and 25 percent of the Ribs and rings. Also, in MRP-227, Rev. 1, Core Shroud Assembly (Welded) Item C2.1, “Remaining Axial Welds,” is a new expansion component applicable to plant designs with core shrouds assembled in two vertical sections. The coverage for Item C2.1 is the same as for Item C3.1. In MRP-227-A, the coverage for the axial welds, ribs and rings was “axial welds seams” other than the core shroud reentrant corner welds at the core mid-plane, plus ribs and rings. Although the extent of coverage required has been quantified, no justification is provided for the examination coverages for the remaining axial welds, or the ribs and rings.

Also, in Figure 4-37, it is not clear if the core shroud assembly can be removed from the core support barrel assembly to allow examination of the ribs and rings.

a. For Item C2.1 and 3.1, does 75 percent of the remaining axial weld length for the remaining axial welds mean a minimum of 75 percent of the total accessible plus inaccessible length of these welds must be examined to claim examination credit?
b. Justify the 25 percent sample size for the ribs and rings (Item C3.2).
c. Clarify whether the ribs and rings are accessible for visual examination.

3.4.2 Industry Response

a. For the expansion inspection of the core shroud assembly ribs and rings, the statement “75% of the remaining axial weld length and adjacent base metal as visible from the core side of the shroud other than that already inspected under the primary link,” specifically refers to 75% of the un-inspected weld length that is visible on the core side of the shroud. This is the high fluence side of the weld. This includes both the accessible and inaccessible portions of the weld length on the core side of the shroud plates; though, it is expected that most if not all of the weld length on the core side will be accessible to the EVT-1 examination.

b. Per the response to part C of this question, it has been determined that the ribs and rings for the CE plants with welded core shrouds assembled with full-height core shroud plates are inaccessible given the current inspection technology available. Consistent with other inaccessible Expansion component items in MRP-227, Revision 1, the examination method/frequency and examination coverage of the ribs and rings in Table 4-5 will be modified as shown in the revised table below. Justification based on an evaluation or some other approach would be required if this expansion inspection is triggered from the primary component inspection of the shroud plates (welded assemblies). The evaluation to perform this justification or determine the alternate

approach must be completed by the end of the next refueling outage after the expansion criteria of the primary component are triggered.

c. The ribs and rings are not accessible given current inspection capabilities. Accessibility to this component is limited to the gap between the core shroud top plate and the core support barrel, which is nominally a 3/16” gap. In lieu of an inspection, an evaluation must be completed that justifies that the aging effects of the component do not adversely affect its ability to perform its function, or the component shall be replaced. This is consistent with other MRP-227 components that have been determined to be inaccessible.

Table 4-5 of MRP-227, Revision 1 will be updated as shown below:

Expansion Item	Applicability	Effect (Mechanism)	Primary Link (Note 1)	Examination Method/Frequency (Note 1)	Examination Coverage
Core Shroud Assembly (Welded) C3.1.Remaining axial welds, C3.2.Ribs and rings	Only plants with welded core shrouds assembled with full-height shroud plates	Cracking (IASCC), Aging Management (IE)	C3.Shroud plates of welded core shroud assemblies	Remaining axial welds: Enhanced visual (EVT-1) examination. Re-inspection every 10 years following initial inspection. Ribs and rings: No examination requirements. Justify by evaluation or by replacement.	Remaining axial welds: 75% of the remaining axial weld length and adjacent base metal as visible from the core side of the shroud other than that already inspected under the primary link. ~~25% of the ribs and rings.~~ Ribs and rings: Inaccessible. See Figure ~~4-38~~ 4-37.

3.5 Response to NRC RAI 16

3.5.1 NRC Question

In MRP-227, Rev.1, Table 4-2, several CE Primary components state under "Examination Method/Frequency," "If screening for fatigue cannot be satisfied by plant-specific evaluation, 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." The language for the corresponding components in MRP-227-A for "Examination scope/frequency" stated "If fatigue life cannot be demonstrated by time-limited aging analysis (TLAA), 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."

The components subject to the fatigue screening are C7., "Core Support Barrel Assembly – CSB [Core Support Barrel] Flexure Weld (CSBFW)," C9., "Lower Support Structure – Core Support Plate," and C10., "Upper Internals Assembly – Fuel Alignment Plate." Also, in Table 4.2, for Item C7., SCC has been added as a degradation mechanism yet the examination method allows examination to be avoided provided the item passes a screening for fatigue.

Therefore, the NRC staff requests that EPRI:

a. Define and justify the criteria that are to be used for screening for fatigue. Is a specific cumulative usage factor (CUF) value used as a screening criterion? Are environmental effects to be considered? If so, how are environmental effects to be included in the evaluation? EPRI should also discuss whether such a criterion should be added to Table 4-2.

b. Justify how fatigue screening accounts for possible SCC contributions for Item C.7? Is additional evaluation or inspection of the CSBFW needed to address possible SCC?

3.5.2 Industry Response

a. The fatigue screening criterion that is provided in MRP-175, Revision 0 [11], which was used in the development of MRP-227-A, is applied for screening for fatigue. MRP-175, Revision 0 utilizes a screening CUF of 0.1 at 40 years, which was intended to address potential environmental effects. Since environmental effects were considered in the MRP-175 screening, there is not a need to add a separate criterion related to it in Table 4-2 of MRP-227, Revision 1.

b. Regarding the degradation mechanisms of the core support barrel flexure weld (CSBFW), fatigue screening does not account for possible contributions from SCC. Provided that the component does not screen-in for fatigue, an inspection would need to be performed to confirm there is no material degradation resulting from SCC. Otherwise, an evaluation could be performed, using plant-specific or bounding information, in place of inspecting the CSBFW for effects of SCC. MRP-227, Revision 1 will be updated as shown below to reflect this item.

Table 4-2 of MRP-227, Rev. 1 is updated as follows:

Primary Item	Applicability	Effect (Mechanism)	Expansion Link (Note 1)	Examination Method/Frequency (Note 1)	Examination Coverage
C7.Core Support Barrel Assembly CSB Flexure Weld (CSBFW)	All plants with welded core shrouds	Cracking (Fatigue, SCC)	None	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.	Examination coverage to be defined by evaluation to determine the potential location and extent of fatigue cracking. See Figure 4-30.

May 17, 2018

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SUBJECT: Transmit Supplemental Information Regarding EPRI Technical Report MRP-227-Revision 1, "*Materials Reliability Program: Pressurized Water Reactor Internals Inspection and Evaluation Guidelines*"

Dear Sir:

This letter is being transmitted to the Nuclear Regulatory Commission (NRC) to support the current NRC safety evaluation of the MRP-227, Revision 1 Reactor Vessel Internals Inspection and Evaluation Guidelines. This letter transmits in Enclosures 1 and 2 responses to NRC staff's supplemental questions in support of the Nuclear Regulatory Commission Safety Evaluation of MRP-227, Revision 1, as was discussed during public meetings on 2/15/2018 and 4/23/2018 with NRC staff, and requested by NRC in ML18053A058 and ML18096A448.

This letter also transmits in Enclosure 3 a comparison of boiling water reactors (BWRs) and pressurized water reactors (PWRs). The focus of this comparison is the experience with stress corrosion cracking (SCC) of austenitic stainless steel alloys in the primary coolant loop of each reactor design and the fundamental chemistry differences that contribute to those differences.

We note that industry is currently reviewing recent operating experience (OE) and MRP-227-A-related inspection findings and this response does not contain any results of that review.

If you have any questions, please contact either Tim Wells-Southern Nuclear Co. (tgwells@southernco.com, 205-992-7460), Kyle Amberge-EPRI-MRP (kamberge@epri.com, 980-266-2646), or myself (mhoehn@ameren.com, (314) 225-1543).

Sincerely,

Mike Hoehn II, Ameren-Missouri
MRP Research Integration Committee Chair

Brian Burgos, Program Manager
EPRI-Materials Reliability Program

Project: 0669
(D035-NRR)

Enclosure 1: Responses to Supplemental Questions Related to WEC/CE PWR Plant Designs in Support of the Nuclear Regulatory Commission Safety Evaluation of MRP-227, Revision 1 (includes two (2) attachments)

Enclosure 2: Responses to Supplemental Questions Related to B&W PWR Plant Designs in Support of the Nuclear Regulatory Commission Safety Evaluation of MRP-227, Revision 1

Enclosure 3: Comparison of Boiling Water Reactor and Pressurized Water Reactor Experience with Cracking of Austenitic Stainless Steel

Reference:
Materials Reliability Program: Pressurized Water Reactor Internals Inspection and Evaluation Guidelines (MRP-227, Revision 1). EPRI, Palo Alto, CA: 2015. 3002005349.

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Responses to Supplemental Questions in Support of the Nuclear Regulatory Commission Safety Evaluation of MRP-227, Revision 1

Prepared for EPRI by:

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Ref:
Westinghouse Electric Co. LTR-AMLR-18-19, Revision 2, dated 4/27/2018

Summary

MRP-227, Revision 1 was completed in 2015 [1] and transmitted to the Nuclear Regulatory Commission (NRC) staff for safety evaluation on December 21, 2015 [2]. Formal requests for additional information (RAIs) were received from the NRC by a letter dated May 15, 2017 [3]. Formal responses to these RAIs were transmitted in two letters transmitted to the NRC staff on October 16, 2017 [4] and January 30, 2018 [5]. The NRC staff sent a set of supplemental questions to the original RAIs to the Electric Power Research Institute (EPRI) in preparation for a public meeting between the staff and industry representatives on February 15, 2018 [6] [7]. The industry presented the details of responses to these supplemental questions at the meeting [8] [9] and followed up the meeting with additional questions and request for transmittal of supplemental information supporting the RAI discussion [10].

This Enclosure transmits two attachments. Attachment 1 includes direct responses to the questions that were transmitted prior to the February 15, 2018 public meeting [6] along with the clarifications provided in the email request following the public meeting [10]. Attachment 1 also includes some small clarifying revisions to the responses to RAIs 8 and 19 previously provided in letter MRP 2017-027 [4]. Attachment 2 provides the additional clarifying information that was requested in response to the meeting through [10]. The contents of this letter can be used by the staff as a reference for developing the safety evaluation on MRP-227, Revision 1.

<u>References</u>:

1. Materials Reliability Program: Pressurized Water Reactor Internals Inspection and Evaluation Guidelines (MRP-227, Revision 1). EPRI, Palo Alto, CA: 2015. 3002005349.
2. Materials Reliability Program Letter, MRP 2015-040, “Report Transmittal: Materials Reliability Program: Pressurized Water Reactor Internals Inspection and Evaluation Guideline (MRP-227 Revision 1), EPRI, Palo Alto, CA, 2015, 3002005349. Ref.: EPRI Project Number 689,” December 21, 2015. (ML15358A046)
3. Nuclear Regulatory Commission Document, “Request for Additional Information for Electric Power Research Institute Topical Report MRP-227, Revision 1, ‘Materials Reliability Program: Pressurized Water Reactor Internals Inspection and Evaluation Guideline’ (CAC No. MF7740),” May 15, 2017. (ML17079A027)
4. Materials Reliability Program Letter, MRP 2017-027, “Responses to NRC Request for Additional Information for Electric Power Research Institute Topical Report MRP-227, Revision 1, ‘Materials Reliability Program: Pressurized Water Reactor Internals Inspection and Evaluations Guideline’ (CAC No. MF7740),” October 16, 2017. (ML17305A056)
5. Materials Reliability Program Letter, MRP 2018-003, “Responses to NRC Requests for Additional Information for Electric Power Research Institute Topical Report MRP-227, Revision 1, “Materials Reliability Program: Pressurized Water Reactor Internals Inspection and Evaluations Guideline” (CAC No. M7740),” January 30, 2018. (ML18038A875)
6. Nuclear Regulatory Commission Document, “Questions for Discussion during February 15, 2018, Nuclear Regulatory Commission Staff and Industry Meeting – MRP-227, Rev. 1 Requests for Additional Information,” February 8, 2018. (ML18038A001)
7. Nuclear Regulatory Commission Document, “2/13/2018 – 2/15/2018 Meeting Agenda on MRP-175 and MRP-211 Revisions and MRP-227, Revision 1 RAI Responses,” January 26, 2018. (ML18024A217)
8. Nuclear Regulatory Commission Document, “Summary of February 13, 2018 – February 15, 2018, Meeting Between the U.S. Nuclear Regulatory Commission Staff and the Industry to Discuss MRP-175, Revision 1, MRP-211, Revision 1 and the Requests for Additional Information on MRP-227, Revision 1,” February 28, 2018. (ML18025B454)
9. Materials Reliability Program Presentation, “MRP-227, Revision 1 Requests for Additional Information,” NRC Public Meeting, Rockville, MD, February 15, 2018. (ML18043B155)
10. Email from Joseph Holonich (NRC) to Kyle Amberge (EPRI), Subject: “Information We Want on the Docket from MRP-227, Rev. 1 RAI Discussion,” February 22, 2018, (ML18053A058)

Attachment 1: Responses to Supplemental Questions in Support of the Nuclear Regulatory Commission Safety Evaluation of MRP-227, Revision 1

Question 1: Response to Request for Additional Information (RAI) 5 in part discussed functionality considerations in support of the reduced sample size for the core barrel welds. Specifically, a core barrel weld could completely fracture allowing the core to drop, and the reactor could still be safely shut down. However, MRP-191 failure modes, effects and criticality analysis (FMECA) categorizes the core barrel welds as high consequence of failure, because failure of these welds could preclude safe shutdown. Please discuss the apparent inconsistency between the RAI 5 response and the MRP-191 FMECA results. [1]

Response:

This apparent inconsistency stems from two aspects of how the MRP-191, Revision 0 [2] and Revision 1 [3] FMECA assessed the core barrel welds:

1. The FMECA combined the economic and safety consequences into one category labeled "Likelihood of Damage". This can be seen in Table 6-3 of either revision of the document.
2. The expert panel at the time assigned a relatively conservative level to the "Likelihood of Failure" category. There were multiple degradation mechanisms and little experience with conducting detailed examinations of the core barrel welds at the time, so it was conservatively assigned a medium likelihood.

The results for the Westinghouse core barrel welds FMECA ranking and categorization from MRP-191, Revision 1 are provided in Table 1. Similar results for the Combustion Engineering (CE) core support barrel welds are provided in Table 2.

The industry is currently in the process of revising MRP-191 for subsequent license renewal (SLR). The same basic expert panel review and FMECA approach as used for earlier revisions of MRP-191 is being used for Revision 2. However, one key change that has been made is to separate the consequences evaluation into economic consequences and safety consequences. Past evaluations documented in the issue management table, MRP-156 [4], defined significant economic impact events for a component as "those for which we do not have a proven fix and would result in significant regulatory and/or public scrutiny, such as first-of-a-kind consideration." The SLR expert panel considered the economic impact of developing solutions and addressing scrutiny, as well, and distilled the economic impact into a ranking based on the expected order of magnitude cost of degradation in a particular component.

Another key difference is the amount of experience accumulated for the core barrel welds since the original publication of MRP-191, Revision 0. This experience ranged from the detailed MRP-227-A [5] enhanced visual examinations already detailed in the response to RAI 5 [6] to the multiple acceptance criteria calculations developed on a plant-specific basis under the methodologies of WCAP-17096-NP-A [7] to the evaluations of the potential for cold work in austenitic stainless steel components in the reactor vessel internals [8].

Publication of MRP-191, Revision 2 is planned for later in 2018, but the expert panel FMECA ranking and categorization review supporting the revised document has already been conducted [9]. Note that for MRP-191, Revision 2 the core barrel welds were separated into girth welds and axial welds because the ranking and categorization is different due to the lower stresses and reduced consequences from postulated degradation (both safety and economic) in the axial welds.

The MRP-191, Revision 2 expert panel conducted extended discussions of the Westinghouse and CE-designed core barrels and the potential for degradation of the welds in the barrels. These discussions included past operating experience and evaluations and consideration of the function of the core barrel. The key points of those discussions are listed here and are separated between lower and upper core barrel and girth and axial welds [9]:

Lower Core Barrel (Westinghouse) or Lower Cylinder (CE) Girth Welds:

- Likelihood of degradation: Low – Welds not expected to fail
  - Thick welds with low active stresses during operation
  - If degradation were to occur it would be expected in the base metal along the weld (heat-affected zone)
  - Full penetration welds – slight concern for fatigue crack growth
  - Small pressure differential across barrel (it is not a pressure vessel)
  - Residual stresses not expected to cause a complete failure
  - Multiple lower core barrel girth welds have been inspected at an EVT-1 level and no relevant indications have been observed to date (see details provided in response to RAI 5 in [6])
  - CE weld operating experience happened early in plant life and was due to a design issue with the attached thermal shield rather than a material aging issue
- Safety consequence of degradation: Medium
  - If core barrel drops, it would be caught by the secondary core support and clevis inserts and radial keys (addressed in more detail in the response to supplemental question 3)
  - Safety consequence assigned to Medium because there is a concern for core damage but the plant could be shut down safely
- Economic consequence of degradation: High
  - Removal of the core barrel after this failure would be a severe challenge
  - Repair may not be possible and replacement of the core barrel would be prohibitively expensive
- Safety category would be A based on the FMECA group, but the panel conservatively elevated it to B
- Economic category is B based on the FMECA group

Lower Core Barrel (Westinghouse) or Lower Cylinder (CE) Axial Welds:

- Likelihood of degradation: Low
  - Considered lower than the likelihood of degradation in girth welds
- Safety consequence of degradation: Low
  - Axial welds do not directly support the core like girth welds
  - Large potential cracks have been justified in acceptance criteria calculations due to low stresses

- Economic consequence of degradation: Medium
    - Ability to justify a large potential crack reduces the economic consequences
    - Economic consequence is medium because actions would likely be required but they would not necessarily lead to extreme cost options

Lower Core Barrel (Westinghouse) or Lower Cylinder (CE) Axial Welds (cont.):
- Safety category is A based on the FMECA group
- Economic category is A based on the FMECA group

Upper Core Barrel (Westinghouse) or Upper Cylinder (CE) Girth Welds:
- Likelihood of degradation: Low – Welds not expected to fail
    - Thick welds with low active stresses during operation
    - If degradation were to occur it would be expected in the base metal along the weld (heat-affected zone)
    - Full penetration welds – slight concern for fatigue crack growth
    - Small pressure differential across barrel (it is not a pressure vessel)
    - Residual stresses not expected to cause a complete failure
    - At least 10-15 upper flange welds (combining both WEC and CE inspections) and several upper girth welds have been inspected with no relevant findings.
        - EVT-1 inspection achieved close to 100% coverage for most of these inspections.
        - See details provided in response to RAI 5 in [6]
    - Little stress corrosion cracking (SCC) has been observed in internals components fabricated from austenitic stainless steel and its weld metals, in general
    - CE weld operating experience happened early in life and was due to a design issue rather than a material aging issue
- Safety consequence of degradation: Medium
    - If core barrel drops, it would be caught by the secondary core support and clevis inserts and radial keys (addressed in more detail in the response to supplemental question 3)
    - Safety consequence assigned to Medium because there is a concern for core damage but the plant could be shut down safely
    - Panel believed that these might be the most fail-safe reactor internals component
- Economic consequence of degradation: High
    - Removal of the core barrel after this failure would be a severe challenge
    - Repair may not be possible and replacement of the core barrel would be prohibitively expensive
- Safety category would be A based on the FMECA group, but the panel conservatively elevated it to B
- Economic category is B based on the FMECA group

Upper Core Barrel (Westinghouse) or Upper Cylinder (CE) Axial Welds:
- (Same discussion as Lower core barrel axial welds)
- Likelihood of degradation: Low
    - Considered lower than the likelihood of degradation in girth welds due to lower active stress during normal operation

Upper Core Barrel (Westinghouse) or Upper Cylinder (CE) Axial Welds (cont.):

- Safety consequence of degradation: Low
  - Axial welds do not directly support the core like girth welds
  - Large potential cracks have been justified in acceptance criteria calculations due to low stresses
- Economic consequence of degradation: Medium
  - Ability to justify a large potential crack reduces the economic consequences
  - Economic consequence is medium because actions would likely be required but they would not necessarily lead to extreme cost options
- Safety category is A based on the FMECA group
- Economic category is A based on the FMECA group

These evaluations by the expert panel resulted in the preliminary results for the Westinghouse core barrel welds provided in Table 3 and the preliminary results for the CE core support barrel welds provided in Table 4 [9]. MRP-191, Revision 2 has not yet been published, so the evaluation results presented above must be called "preliminary." It should be noted that this expert panel evaluation was conducted prior to the core barrel operating experience gained during the spring 2018 outage season. This recent operating experience at a Combustion Engineering-designed plant may have an impact on the likelihood of degradation rankings provided here, but the technical discussions supporting the safety consequence rankings should not be affected.

Table 3 and Table 4 show two key points that resolve the inconsistency noted between the RAI 5 response and the MRP-191 FMECA results:

1. When economic consequence and safety consequence are separated, the safety consequence is medium, while the economic consequence is high. The medium safety consequence is due to the same reasoning about core shutdown still being possible, which has been detailed in the response to RAI 5 [6]. The high economic consequence is due to the possibility of core barrel degradation requiring a replacement or costly repairs.
2. The likelihood of degradation was reduced to medium based on the extensive operating experience from MRP-227-A inspections and the significantly improved understanding of stress in the core barrel and its welds.

The FMECA ranking and categorization results from the MRP-191, Revision 2 expert panel are consistent with the technical basis provided in the response to RAI 5 [6] and are due to the increased experience and better technical understanding of the welds and core barrel.

Table 1: MRP-191, Revision 1 FMECA Ranking and Categorization Table for the Westinghouse Core Barrel Welds

Assembly	Subassembly	Component	Material	Screened-in Degradation Mechanisms	Likelihood of Failure L, M, H	Likelihood of Damage L, M, H	FMECA Group	Category
Lower internals assembly	Core barrel	Lower core barrel (includes LGW, LFW, MAW, and LAW)	304 SS	Weld, IASCC, IE	M	H	3	C
		Upper core barrel (includes UFW, UGW, and UAW)	304 SS	Weld, IE	M	H	3	C

Table 2: MRP-191, Revision 1 FMECA Ranking and Categorization Table for the Combustion Engineering Core Support Barrel Welds

Assembly/ Subassembly	Component	Material	Screened-in Degradation Mechanisms	Likelihood of Failure L, M, H	Likelihood of Damage L, M, H	FMECA Group	Category
Core Support Barrel Assembly	Upper cylinder (includes UFW, UGW, and UAW)	304 SS	Weld	L	H	2	B
	Lower cylinder (includes MGW, LGW/LFW, MAW, LAW, and CSBFW)	304 SS	Weld, IASCC, IE	M	H	3	C

Table 3: MRP-191, Revision 2 (SLR) FMECA Ranking and Categorization Table for the Westinghouse Core Barrel Welds

Assembly	Subassembly	Component	Material	Screened-in Degradation Mechanisms	Likelihood of Failure	Safety Consequence	Economic Consequence	Safety FMECA Group	Economic FMECA Group	Safety Category	Economic Category
Lower Internals Assembly	Core barrel	Lower core barrel axial welds (includes middle axial weld (MAW) and lower axial weld (LAW))	304 SS	Weld, IASCC, Fatigue, IE, VS	L	L	M	1	1	A	A
		Lower core barrel girth welds (includes lower girth weld (LGW) and lower flange weld (LFW))	304 SS	Weld, IASCC, Fatigue, IE, VS	L	M	H	1	2	B	B
		Upper core barrel axial welds (includes upper axial weld (UAW))	304 SS	Weld, Fatigue	L	L	M	1	1	A	A
		Upper core barrel girth welds (includes upper flange weld (UFW) and upper girth weld (UGW))	304 SS	Weld, Fatigue	L	M	H	1	2	B	B

Table 4: MRP-191, Revision 2 (SLR) FMECA Ranking and Categorization Table for the Combustion Engineering Core Support Barrel Welds

Assembly	Component	Material	Screened-in Degradation Mechanisms	Likelihood of Failure	Safety Consequence	Economic Consequence	Safety FMECA Group	Economic FMECA Group	Safety Category	Economic Category
Core Support Barrel assembly	Upper cylinder girth welds (Upper Flange Weld, Upper Girth Weld)	304 SS	Weld, IASCC, Fatigue, IE	L	M	H	1	2	B	B
	Upper cylinder axial welds (Upper Axial Weld)	304 SS	Weld, IASCC, Fatigue, IE	L	L	M	1	1	A	A
	Lower cylinder girth welds (Middle Girth Weld, Lower Girth Weld /Lower Flange Weld)	304 SS	Weld, IASCC, Fatigue, IE	L	M	H	1	2	B	B
	Lower cylinder axial welds (Middle Axial Weld, Lower Axial Weld)	304 SS	Weld, IASCC, Fatigue, IE	L	L	M	1	1	A	A

Question 2: RAI 5, Tables 1-3, what is the probability of detection for each crack size? How do you get >25% probability when inspecting a 25% sample with only one crack present? (Maybe provide an example of one of these calculations). [1]

Response:
"Probability of Detection" refers to two aspects of this particular question and response. The first aspect is the direct subject of RAI 5 [6], which provides information on the probability of detecting one or more cracks of a specific size in a typical core barrel weld length during an EVT-1 inspection with 25% coverage. The tables apply to either a single inspection with one or more cracks present of that particular size or to multiple welds with one or more cracks of that size.

The second aspect of probability of detection is the probability that a particular EVT-1 inspection will detect a crack of a certain size. This is implicitly included in the response to RAI 5 [6] in the assumptions about minimum detectable crack size (0.25 inch in the response) and how it is treated in the calculations. Cracks below 0.25 inch are assumed to have zero probability of detection, which is likely conservative. Cracks 0.25 inch and longer are assumed to have 100% probability of detection. For longer cracks, this assumption is expected to be close to the real probability of detection. Smaller cracks may be more difficult to detect, but the EVT-1 requirements of the inspection standard, MRP-228 [10], are intended to maximize the probability of detection of the inspection technique. These include but are not limited to the demonstration, cleanliness, travel speed, angle, distance, and lighting requirements. A round robin including several inspection vendors was conducted to evaluate the effectiveness of the required EVT-1 inspections in detecting cracks of various sizes and locations [11]. The round robin results showed that with a crack length of 6-10 mm (0.24-0.39 inches) the average detection rate was 62%. This rose to 88% for cracks in the 16-20 mm (0.63 – 0.79 inches) range. The round robin demonstrated that probability of detection is vendor and inspection system specific, so these detection rates were not included in the calculations supporting response to RAI 5 beyond the assumption of a minimum detectable crack size.

The probability of detecting a crack within a length of weld in these calculations is dependent on the size of the crack. As noted in the response to RAI 5:

> *Assumed crack size has a slight effect on the probability of detection, with larger cracks having a higher probability than smaller cracks. In this calculation, the increase in probability comes from the potential to detect the end of a crack that extends into the uninspected portion of the weld.*

This is due to the assumption that 0.25 inch of crack length intersecting the inspection length will be detected, while any length less than 0.25 inch will not be detected. For a 0.25 inch crack, this means that if the crack is fully present in the inspection length, it will be detected, but if even 0.000000001 inch of the crack is outside of the assumed inspection length, it will not be detected. This is demonstrated in Figure 1 for an assumed 0.25 inch flaw. If the 0.25 inch flaw is fully within the inspected length, then it is detected. If even a small portion of it is outside of the inspection length, it is assumed to be missed. This is conservative, because there are practical limits to what would be missed.

Thus, for a 0.25 inch crack:

Probability = inspection coverage (25%) – 0.25"/weld length = 24.9%.

The same logic applies to a longer crack. Figure 2 demonstrates a similar case for a longer flaw. If at least 0.25 inch of the flaw intersects the inspected length, the flaw will be detected. However, if less than 0.25 inch of the flaw intersects; it is assumed to be undetected. Thus, for larger cracks, the inspected length is effectively increased by the assumed crack length minus the minimum detectable size to account for a crack that just intersects the end of the inspected length. For a single 2 inch crack in a weld, this increases the inspected length effectively by 1.75 inch.

For a 2 inch crack:

Probability = inspection coverage (25%)+(crack length-0.25")/weld length-0.25"/weld length = 25.3%

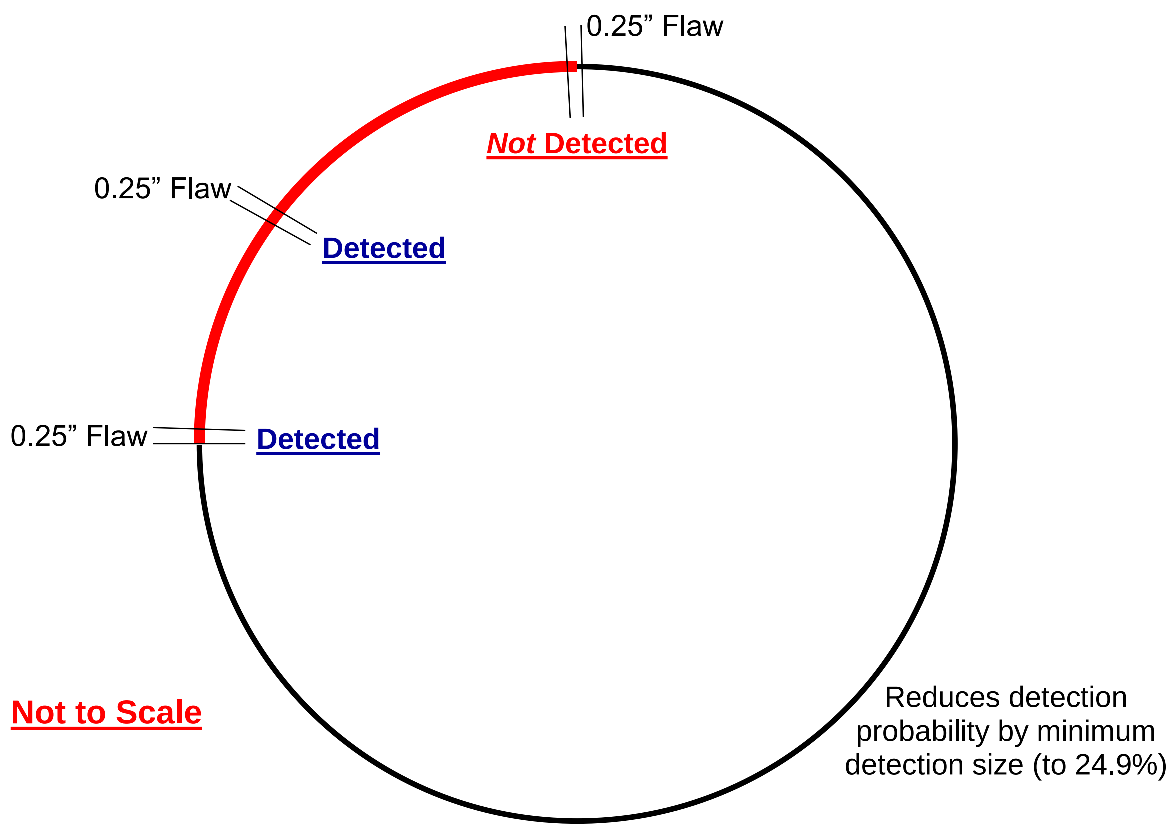

Figure 1: Schematic of core barrel weld inspection coverage and flaw intersection for detection by a 25% EVT-1 sampling inspection (0.25 inch flaw)

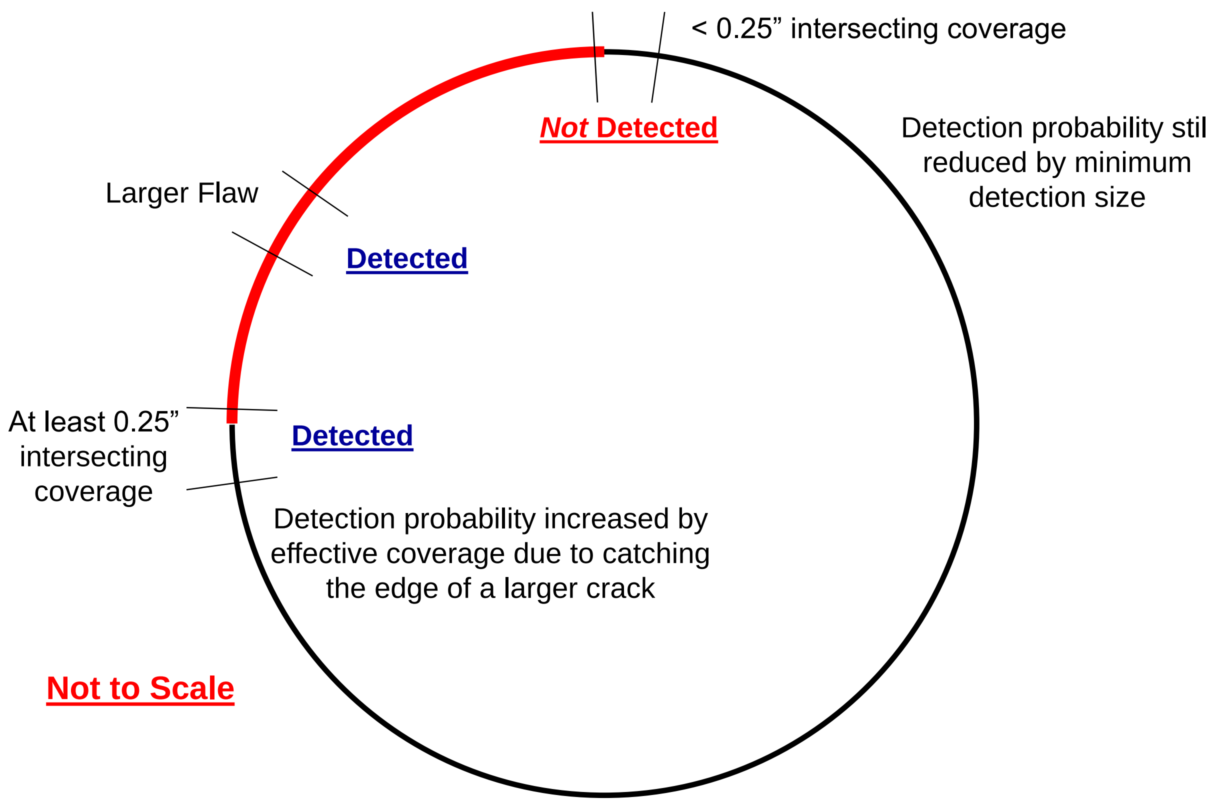

Figure 2: Schematic of core barrel weld inspection coverage and flaw intersection for detection by a 25% EVT-1 sampling inspection (flaw sizes greater than 0.25 inch)

Question 3: RAI-5—Response from the Material Reliability Program (related to the functionality of the core barrel under a faulted condition) states in part the following—Page 11 of LTR-AMLR-17-9, Rev.2:

"Testing was conducted to measure the effect of various abnormal conditions on the ability to insert the control rods and the time to scram. One of these tests investigated the effect of a full core drop type accident.

The testing performed ... also tested the effect of significant fuel deflections (i.e., the center of the fuel assembly was deflected laterally while the top and bottom were pinned) and determined that effect on scram time was acceptable. This provides evidence that the small 'bend' in the control rod insertion path that could be caused by a tilted core barrel would not have an impact on the ability to insert the control rods for core shutdown."

The staff requests industry discuss the following:

- During the scenario addressed above, how many control rod assemblies are allowed to encounter the small "bend" in the control rod insertion path due to a tilted core barrel?
- Is the deflection due to small "bend" observed (in the testing) in the control rod insertion path bounded by the safety margin established in a plant loss-of-coolant-accident accident analyses for each Combustion Engineering and Westinghouse unit?
- Provide a brief summary on how the full core drop test was conducted. [1]

Response:

Overview of Core Girth Weld Failure

Some additional background information is required prior to describing these test conditions and results. One of the purposes of the control rod insertion tests was to simulate the effects of a core barrel failure on rod insertion times and the capability for full rod insertion. This requires an understanding of what will occur during a hypothetical core barrel failure. Figure 3 contains Figure 3-5 from MRP-227, Revision 1 [12], which shows a Westinghouse-designed reactor vessel with the reactor vessel internals inside. Key components that are relevant to understanding what occurs if a core barrel fails include:

- the core barrel itself
- the core barrel outlet nozzles and interfacing vessel outlet nozzles
- the radial keys and interfacing clevis inserts
- the secondary core support (SCS) structure
- the upper core plate alignment pins

The components attached directly to the core barrel are shown in more detail in Figure 4.

Several scenarios for failure of a core barrel girth weld can be considered. Each of these will result in some interaction between the alignment and interfacing components listed above.

- Girth weld above the baffle-former assembly (core shroud assembly in CE plants) fails
  - Complete 360° failure
  - Partial failure
- Girth weld within or below the baffle-former assembly (core shroud assembly in CE plants) fails
  - Complete 360° failure
  - Partial failure

Complete, 360° failure of a girth weld above the baffle-former assembly would likely result in the entire lower internals dropping down. The SCS structure is specifically designed to limit the vertical displacement of the lower internals after a postulated core barrel girth weld failure at the core barrel to flange weld. Should this postulated failure occur, the drop of the core barrel would cause the bottom plate of the SCS to displace vertically by a small amount and come into contact with the bottom head of the reactor vessel. The energy absorbers are designed to absorb the impact from this drop and prevent damage. The maximum possible drop from a complete girth weld failure is less than the distance the control rods are inserted into the fuel during normal operation, so they will stay aligned with the fuel assemblies and ready for insertion to achieve safe shutdown. The maximum possible drop is also less than the length of the fuel alignment pins, which prevents the top of the fuel assembly from shifting out of alignment with the control rod guide tube assembly and the control rods. Lateral support to the barrel as a whole during this hypothetical complete girth weld failure would be provided by the core barrel outlet nozzles, the radial keys, and the fuel alignment pins. The outlet nozzles are nearly in contact with the vessel outlet nozzles during normal operation, and would not allow significant tilting or shifting of the barrel near the top. The radial keys have a small gap interface with the clevises on the vessel and provide lateral and rotational support to the barrel, also limiting the potential for offset or twist in the barrel.

A partial failure of a girth weld above the baffle-former assembly could result in one side of the barrel being tilted downwards while the other side is still partially attached to the upper portion of the barrel. The vertical drop distance on the fractured side is still limited by the SCS, and any tilt would be limited by the radial keys. The same discussion of the potential vertical and lateral offsets of the fuel assemblies and control rods discussed for a complete failure still apply. The interfaces of the outlet nozzles and the radial keys will also limit the potential offsets and bends that can occur in this case.

This first group could be further divided into failures above and below the core barrel outlet nozzles. However, the SCS, the radial keys, and the engagement of the fuel alignment pins will limit potential lateral or circumferential movement in all cases.

Girth weld failures below the top of the baffle-former assembly present even less of a concern due to the presence of the other lower internals components attached to the core barrel. All of the discussion about the constraints provided by the SCS and radial keys for failures above the baffle-former assembly still apply, but the baffle-former assembly structure and other components like the lower support columns will act as a secondary support to reduce the potential lateral and vertical offsets. Thus, any test or argument that addresses a failure in the upper part of the core barrel will address failures in the lower part.

Note that CE-designed core support barrels have components that perform the same functions as described above for the Westinghouse-designed case. These components typically have different names, often look a bit different, and in some cases were placed in a different location, but the reasoning presented above is equally applicable to CE-designed internals.

Description of Control Rod Insertion Test (Response to Question Part 3)

Since a brief summary of how the test was conducted (question in bullet 3) is pertinent in responding to the other two questions, this is addressed next. The overall purpose of the test conducted was to evaluate how deflections at various points in the drive line could impact control rod insertion times. Various tests were conducted during the test program; however, only the two tests relevant to the questions asked are discussed.

The test setup was comprised of a simulated control rod drive mechanism (CRDM) and related driveline components, a full size prototypic 17x17 standard guide tube assembly, and a full size prototypic 17x17 fuel assembly [13]. The fuel assembly was submerged in water and the guide tube was wetted in an attempt to achieve a prototypic friction between the control rods and guide tube and fuel assembly. The simulated CRDM contained features which allowed for tuning of drop times to match drop times achieve during full-flow loop drop tests. The drop times were also confirmed prior to performing any tests to be repeatable within less than ± 1-percent.

The first test was conducted considering a vertical and lateral offset of the fuel assembly top nozzle [13]. The purpose of this test was to assess the impact that a core drop would have on rod insertion times. The vertical offset considered in the test is on the order of and slightly greater than the vertical offset expected for the core drop condition. Similarly, the lateral offset simulated was representative of the lateral offset that would occur between the fuel assembly top nozzle and the fuel alignment pin when the fuel top nozzle was vertically offset due to the core drop. The results of this test show less than an 8-percent impact on rod insertion times in the configuration described and that the rods were able to fully insert.

The second test was conducted considering a lateral offset of the fuel assembly at mid span with the top and bottom of the fuel assembly pinned [13]. The purpose of this test was to assess the impact that a fuel assembly deflections would have on rod insertion times. The lateral offset used for this test was representative of approximately one-half of the cumulative gap between fuel assemblies at the widest width of the core for a plant that uses 17x17 fuel assemblies. In other words, it assumed a case where half of the fuel assemblies in the core were deflected to one side such that the grids nearest to the mid-span of the fuel assembly were in contact across the core and the grid on the peripheral fuel assembly is in contact with the baffle plate. The results of this test show less than a 2-percent impact on rod insertion times in the configuration described and that the rods were able to fully insert.

Response to Question Part 1

With regards to the first part of this question (bullet 1), only one path was simulated by the test. In the case of a core drop, it would be expected that all fuel assemblies would be laterally offset by a similar amount and have some slight variation in vertical offset associate with the slight tilt of the barrel prior to contact at the outlet nozzle gaps. Therefore, the results of this test are expected to be representative of all control rod locations.

Response to Question Part 2

With regards to the second part of this question (bullet 2), Westinghouse has not performed a review of deflections calculated for all Westinghouse and CE loss-of-coolant accident analyses to confirm that the deflections simulated in the test are bounding of these analyses. However, it is expected that the 17x17 test adequately represents all of the other applicable fuel designs. This is based on the testing parameters and results, specifically, the significant deflections applied relative to the total available gap across the core, the nearly negligible impact on rod insertion times, and the ability to fully insert the rods. It is expected that a more expansive search would result in showing that the test results are representative or bounding.

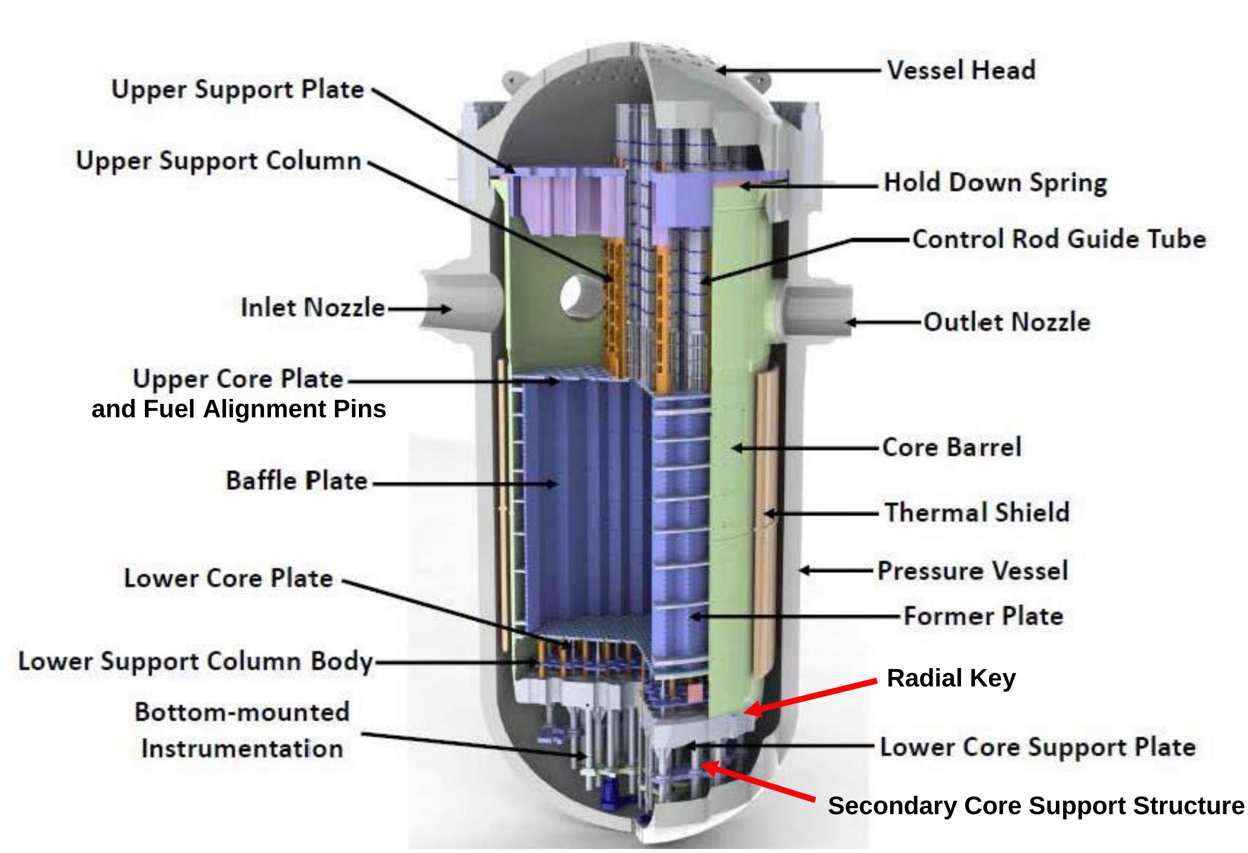

Figure 3: Westinghouse-design reactor vessel and reactor vessel internals ([12] Figure 3-5)

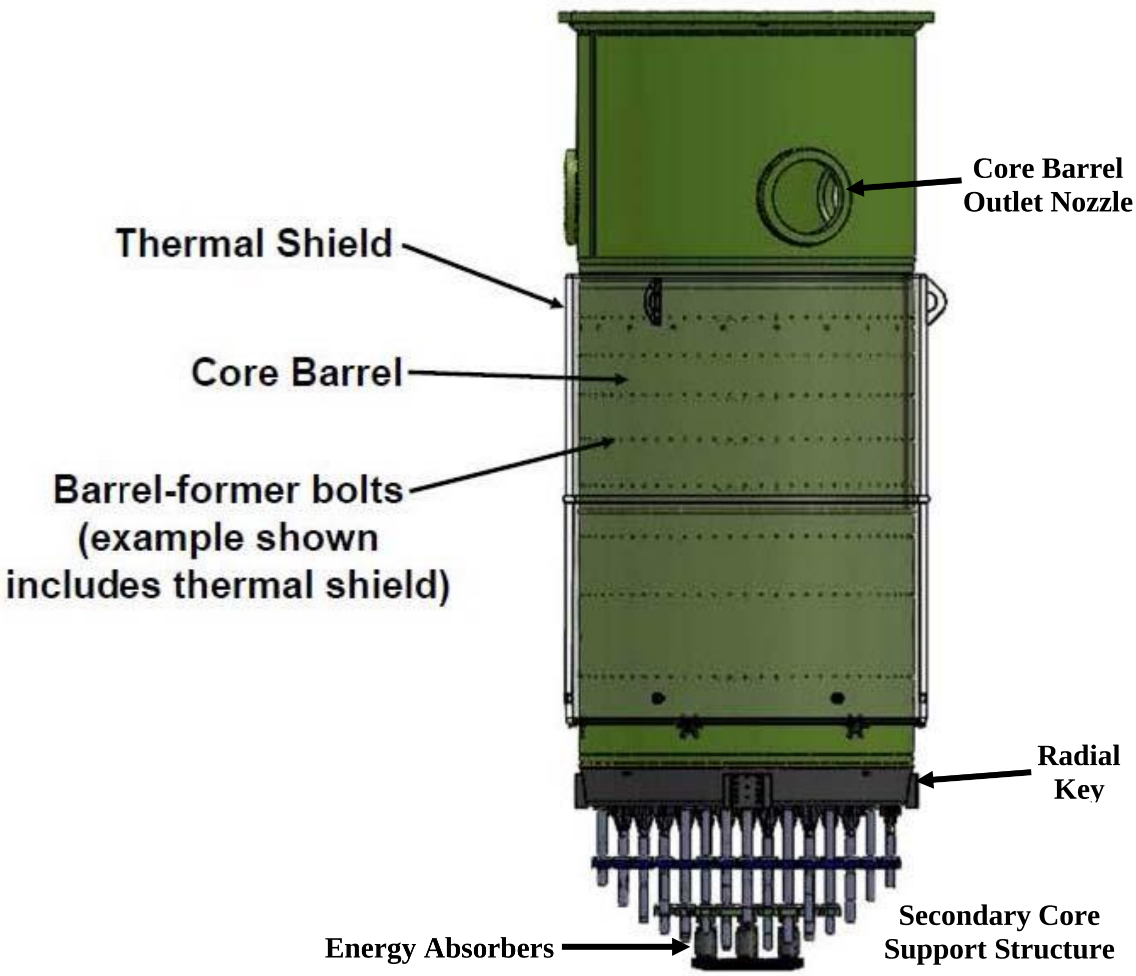

Figure 4: Westinghouse-design core barrel and secondary core support structure ([12] Figure 4-21)

Question 4: The response to RAI 10, which concerns the adequacy of a 25% sample inspection of the deep beam welds, provides a markup of Table 5-2 showing the expansion to the remaining deep beam welds if cracking is found in the initial sample. The markup shows this expansion inspection must be completed by the end of the next refueling outage.

Why not require the expansion be completed during the same refueling outage during which the cracking was found in the initial sample, consistent with the approach for the core barrel welds? [1]

Response:

The technical basis for allowing one cycle in which to complete the inspection is the difference in redundancy and function of an individual weld between the deep beams and the core barrel. Each beam is kept in place by more than one weld, and the failure of an entire weld would not result in loss of functionality. This is quite different from the impact of a core barrel girth weld failure. Additionally, the insertion and removal of fuel each outage provides an element of regular monitoring to the deep beams. Both of these elements are described in the response to RAI 10 [6]:

> *The function of the deep beams is to directly support the core, to keep the fuel in place and to maintain alignment for control element assembly insertion. From the standpoint of functionality, the welded array is a redundant structure. If one weld of a cross-beam fails completely, the other end of that particular beam would still be attached to another main beam. The main beams are welded at multiple locations and would require multiple weld failures to compromise function. Assurance of the continued functionality of the deep beams is also aided by the fact that the onset of the loss of structural functionality would be likely to be first detected during fuel loading or unloading conducted during each refueling outage. The fuel loading and unloading operations are expected to detect this loss of functionality as misaligned fuel assemblies or abnormal difficulty with removing or placing fuel assemblies.*

The additional cycle also has the added benefit of providing the utility with additional flexibility in planning and implementing this expansion inspection most efficiently and effectively, allowing it to be completed during the same outage or the next one.

Clarification on the Response to RAI 8 Provided in Letter MRP 2017-027 [14]

The response to RAI 8 included a note on the determination of re-examination periods for the baffle-former bolts. This was Note 12 and stated:

> *"12. Re-examination periods shall be determined by plant-specific evaluation per the MRP-227 Needed Requirement 7.5 as documented and dispositioned in the owner's plant corrective action program. If atypical or aggressive baffle-former bolt degradation is observed, the interim guidance (MRP 2016-021 and MRP 2017-009) provides limitations to the permitted re-inspection interval (not to exceed 6 years maximum) unless further evaluation is performed to justify a longer interval. If evaluation justifies a longer re-inspection interval, it is not permitted to exceed 10 years."* [14]

The definition of "atypical or aggressive baffle-former bolt degradation" was not explicitly defined in this note. The intention was to use the same definition of "atypical or aggressive baffle-former bolt degradation" used in the baffle-former bolt interim guidance in letter MRP 2017-009 [15]. Thus, Note 12 will be modified as follows:

> "12. Re-examination periods shall be determined by plant-specific evaluation per the MRP-227 Needed Requirement 7.5 as documented and dispositioned in the owner's plant corrective action program. If atypical or aggressive baffle-former bolt degradation as defined in MRP 2017-009 (i.e., ≥3% of baffle-former bolts with UT or visual indications or clustering* for downflow plants and ≥5% of baffle-former bolts with UT or visual indications or clustering* for upflow plants) is observed, the interim guidance (MRP 2016-021 and MRP 2017-009) provides limitations to the permitted re-inspection interval (not to exceed 6 years maximum) unless further evaluation is performed to justify a longer interval. If evaluation justifies a longer re-inspection interval, it is not permitted to exceed 10 years.
> *"Clustering" is defined per NSAL-16-1 Rev.1 as three or more adjacent defective BFBs or more than 40% defective BFBs on the same baffle plate. Untestable bolts should be reviewed on a plant-specific basis consistent with WCAP-17096-NP-A for determination if these should be considered when evaluating clustering."

Clarification 1 on the Response to RAI 9 Provided in Letter MRP 2018-003 [6]

The response to RAI 9 in letter MRP 2018-003 [6] clarified the intention of the coverage requirements for the Westinghouse lower support columns (LSC) and the CE core support columns in MRP-227, Revision 1 and included revised entries for Table 4-5 and Table 4-6. However, these tables did not specify the distribution of the 25% coverage requirement for the LSCs or core support columns. The stress and dose in the columns for both designs are expected to vary from the center of the lower core plate or core support plate to the outer edge of the plate. To address the variation in column degradation behavior that may occur due to these variations in stress and dose, a requirement to evenly distribute the inspection across the population of LSCs or core support columns will be added to these two tables. Note that the CE core support columns were a Primary component in MRP-227, Revision 1 and therefore included in Table 4-2. The response to RAI 9 in [6] provides the basis, and corresponding markup, to move the core support columns from a Primary to an Expansion component, and therefore include in Table 4-5.

For the CE core support columns, the revised text for the "Examination Coverage" column of Table 4-5 will read (showing changes from MRP-227, Revision 1 text):

> "Plants with full height bolted core shroud plates: 25% of the total number of column assemblies (both visible and non-visible from above the core support plate) ~~as visible~~ using a VT-3 examination from above the core support plate. The inspection coverage must be evenly distributed across the population of column assemblies.
>
> Plants with core shrouds assembled in two vertical sections: 25% ~~100%~~ of the accessible surfaces of the core support column welds, from the top side of the core support plate ~~(Note 3)~~. The inspection coverage must be evenly distributed across the population of core support column welds.
>
> (Notes 3 and 4)"

The added Note 4 for Table 4-5 will state:

> "4. Justification that adequate distribution of the inspection coverage has been achieved can be based on geometric or layout arguments. Possible examples include, but are not limited to, inspection of all column assemblies or accessible core support column welds in one quadrant of the core support plate (based on the azimuthal symmetry of the plate) or inspecting every fourth column or weld across the entire plate."

For the Westinghouse LSCs, the revised text for the "Examination Coverage" column of Table 4-6 will read (showing changes from MRP-227, Revision 1 text):

> "25% of the total number of column assemblies (both visible and non-visible from above the lower core plate) using a VT-3 examination from above the lower core plate. The inspection coverage must be evenly distributed across the population of column assemblies.
>
> (Notes 3 and 4)"

The added Note 4 for Table 4-6 will state:

> "4. Justification that adequate distribution of the inspection coverage has been achieved can be based on geometric or layout arguments. Possible examples include, but are not limited to, inspection of all column assemblies in one quadrant of the lower core plate (based on the azimuthal symmetry of the plate) or inspecting every fourth column across the entire plate."

Clarification 2 on the Response to RAI 9 Provided in Letter MRP 2018-003 [6]

The response to RAI 9 in letter MRP 2018-003 [6] states that if degradation is observed in the initial inspection population for either the Westinghouse LSCs or the CE core support columns, then the examination would expand to include the remainder of the population of the column bodies. Note 3 was added to Table 4-5 and Table 4-6:

> "*3. The stated minimum coverage requirement is the minimum if no significant indications are found. However, the Examination Acceptance criteria in Section 5 require that additional coverage must be achieved in the same outage if significant flaws are found. This contingency should be considered for inspection planning purposes*." [6]

This note clearly states that the expansion must be conducted during the same outage in which significant flaws are found. However, the text in Table 5-2 for CE and Table 5-3 for Westinghouse requires some modification to clearly state this timing requirement.

Additionally, Table 5-2 and Table 5-3 need to clearly state the level of degradation required to trigger the expansion from the 25% sample to the remaining LSCs or core support columns. Descriptions of the level of degradation appropriate for triggering the expansion were already provided in the "Expansion Item Examination Acceptance Criteria" column of Table 5-3 of MRP-227, Revision 1 for the Westinghouse LSCs and in the same column for the revised Table 5-2 entry provided in the response to RAI 9 in letter MRP 2018-003.

To improve the clarity of MRP-227 Tables 5-2 and 5-3, text providing the expansion criteria and timing for the LSCs and core support columns will be added to the "Examination Method/Frequency" column of the tables.

The following paragraphs will be added to the "Expansion Criteria" column of Table 5-2 for the CE core support columns:

> "Plants with full height bolted core shroud plates: The confirmed detection of missing or separated welds in a core support column or fractured, misaligned, or missing core support columns shall require examination of 100% of the accessible uninspected core support column assemblies using a VT-3 examination from above the core support plate (minimum of 75% of the total population of core support column assemblies) during the same outage.
>
> Plants with core shrouds assembled in two vertical sections: The confirmed detection of a relevant disruption of discontinuity in the surface of a core support column weld shall require examination of 100% of the accessible uninspected core support column welds from the top side of the core support plate (minimum of 75% of the total population of core support column welds) during the same outage."

The following paragraph will be added to the "Expansion Criteria" column of Table 5-3 for the Westinghouse LSCs:

> "The confirmed detection of fractured, misaligned, or missing lower support columns shall require examination of 100% of the accessible uninspected lower support column assemblies using a VT-3 examination from above the lower core plate (minimum of 75% of the total population of lower support column assemblies) during the same outage."

Clarification on the Response to RAI 19 Provided in Letter MRP 2017-027 [14]

The response to RAI 19 in letter MRP 2017-027 [14] provides a revised entry for the Control Rod Guide Tube Assembly Guide plates (cards) to be used in Table 4-3 of MRP-227, Revision 1. This revised table entry implements the requirements of WCAP-17451-P [16] under both the "Examination Method/Frequency" and "Examination Coverage" columns and references Note 7 to provide more information on WCAP-17451-P:

> *"7. In WCAP-17451-P Revision 1 the baseline examination schedule has been adjusted for various CRGT designs, the extent of individual CRGT examination modified, and flexible subsequent examination regimens correlating to initial baseline sample size, accuracy of wear estimation and examination results. Initial inspection prior to the license renewal period may be required. Refer to the latest revision of WCAP-17451-P including the results of the NEI 03-08 Generic Topical Report screening and/or NRC review for the specific guidance elements."* [14]

The statement in the last sentence of this note implementing the "latest revision of WCAP-17451-P" will be revised to avoid potential issues for the development of a safety evaluation on MRP-227, Revision 1.

The current applicable version of WCAP-17451-P is Revision 1 [16]. This revision is still applicable for many plants. However, recent operating experience has led to the creation of interim guidance on WCAP-17451-P, Revision 1 for certain control rod designs. This interim guidance has been issued as NEI 03-08 "Needed" guidance under Pressurized Water Reactor Owners Group (PWROG) letter OG-18-46 [17] and was provided to the staff for information under Pressurized Water Reactor Owners Group letter OG-18-76 [18]. For MRP-227, Revision 1, Note 7 will be revised to reference WCAP-17451-P, Revision 1 as modified by the interim guidance of letter OG-18-46. The revised note will state:

> "7. In WCAP-17451-P ~~Revision 1~~ the baseline examination schedule has been adjusted for various CRGT designs, the extent of individual CRGT examination modified, and flexible subsequent examination regimens correlating to initial baseline sample size, accuracy of wear estimation and examination results. Initial inspection prior to the license renewal period may be required. ~~Refer to the latest revision of~~ Use WCAP-17451-P, Revision 1, including the modified requirements due to the interim guidance provided in letter OG-18-46. ~~results of the NEI 03-08 Generic Topical Report screening and/or NRC review for the specific guidance elements.~~"

References:

1. Nuclear Regulatory Commission Document, "Questions for Discussion during February 15, 2018, Nuclear Regulatory Commission Staff and Industry Meeting – MRP-227, Rev. 1 Requests for Additional Information," February 8, 2018. (ML18038A001)
2. *Materials Reliability Program: Screening, Categorization, and Ranking of Reactor Internals Components for Westinghouse and Combustion Engineering PWR Design (MRP-191).* EPRI, Palo Alto, CA: 2006. 1013234.
3. *Materials Reliability Program: Screening, Categorization, and Ranking of Reactor Internals Components for Westinghouse and Combustion Engineering PWR Design (MRP-191, Revision 1).* EPRI, Palo Alto, CA: 2016. 3002007960.
4. *Materials Reliability Program: Pressurized Water Reactor Issue Management Table, PWR-IMT Consequence of Failure (MRP-156).* EPRI, Palo Alto, CA: 2005. 1012110.
5. *Materials Reliability Program: Pressurized Water Reactor Internals Inspection and Evaluation Guidelines (MRP-227-A).* EPRI, Palo Alto, CA: 2011. 1022863.
6. Materials Reliability Program Letter, MRP 2018-003, "Responses to NRC Requests for Additional Information for Electric Power Research Institute Topical Report MRP-227, Revision 1, "Materials Reliability Program: Pressurized Water Reactor Internals Inspection and Evaluations Guideline" (CAC No. MF7740)," January 30, 2018. (ML18038A875)
7. Westinghouse Report, WCAP-17096-NP-A, Revision 2, "Reactor Internals Acceptance Criteria Methodology and Data Requirements," August 2016.
8. Pressurized Water Reactor Owners Group Document, PWROG-15105-NP, Revision 0, "PA-MSC-1288 PWR RV Internals Cold-Work Assessment," April 2016.
9. Westinghouse Letter, LTR-AMLR-17-28, Revision 0, "MRP-191 Revision 2 Expert Panel Meeting Minutes," January 22, 2018 (Westinghouse Proprietary).
10. *Materials Reliability Program: Inspection Standard for Pressurized Water Reactor Internals—2015 Update (MRP-228, Rev. 2).* EPRI, Palo Alto, CA: 2015. 3002005386.
11. *Remote Visual Testing Round-Robin Study*. EPRI, Palo Alto, CA: 2016. 3002007793.
12. *Materials Reliability Program: Pressurized Water Reactor Internals Inspection and Evaluation Guidelines (MRP-227, Revision 1)*. EPRI, Palo Alta, CA: 2015. 3002005349.
13. Westinghouse Report, WCAP-9251, "SCRAM Deflection Test Report 17x17 Guide Tubes, 96 Inch and 150 Inch," December 1977 (Westinghouse Proprietary).
14. Materials Reliability Program Letter, MRP 2017-027, "Responses to NRC Request for Additional Information for Electric Power Research Institute Topical Report MRP-227, Revision 1, 'Materials Reliability Program: Pressurized Water Reactor Internals Inspection and Evaluations Guideline' (CAC No. MF7740)," October 16, 2017. (ML17305A056)
15. Materials Reliability Program Letter, MRP 2017-009, "Transmittal of NEI-03-08 "Needed" Interim Guidance Regarding Baffle Former Bolt Inspections for PWR Plants as Defined in Westinghouse NSAL 16-01 Rev. 1," March 15, 2017.
16. Pressurized Water Reactor Owners Group Report, WCAP-17451-P, Revision 1, "Reactor Internals Guide Tube Wear – Westinghouse Domestic Fleet Operational Projections," October 2013.
17. Pressurized Water Reactor Owners Group Letter, OG-18-46, "Transmittal of Approved "Needed" Interim Guidance for Addressing Accelerated Guide Card Wear Issue Described in NSAL-17-1, (LTR-ARIDA-17-270, Revision 0), PA-MSC-1471," February 20, 2018.
18. Pressurized Water Reactor Owners Group Letter, OG-18-76, "Transmittal of Interim Guidance for Addressing Accelerated Guide Card Wear Issue Described in NSAL-17-1 (PA-MSC-1471)," March 23, 2018. (ML18088A199)

Attachment 2: Supplemental Clarification Material Requested by the NRC Staff in Support of the Safety Evaluation of MRP-227, Revision 1

The NRC staff requested further supplemental clarification material based on the industry presentation at the February 15, 2018 [1]. This request was transmitted via email and recorded in the NRC's public document repository [2]. Portions of the supplemental information requested have already been included in the responses provided in Attachment 1 of this letter. The remaining information requested falls into two categories:

- Expected industry response to the observation of degradation in the core barrel welds
- Additional detail on the comparison between boiling water reactor (BWR) and pressurized water reactor (PWR) experience and environmental conditions

The February 15, 2018 meeting included a discussion of the limitations on core barrel weld inspection coverage due to accessibility issues and the serious risks associated with reactor vessel internals disassembly. This discussion has been included here as relevant supplemental information.

Industry Response to Observed Degradation

Observation of relevant indications in the core barrel welds or in any MRP-227-A [3] or Revision 1 [4] inspection component would trigger a multi-part response from the affected licensee. Notification of the industry and the NRC would be included in this response along with evaluation of the extent of condition and evaluation and disposition of the relevant finding.

Section 7 of MRP-227-A [3] and Revision 1 [4] governs the response to relevant findings. Quotes from this section provided here are from MRP-227-A. Section 7.5 “Examination Results Requirement” dictates that examination results which do not meet the acceptance criteria of MRP-227, Section 5 “shall be recorded and entered in the plant corrective action program and dispositioned.” Section 7.6 “Aging Management Program Results Requirement” requires that a summary report of inspection and evaluation experience be provided to the MRP Program Manager within 120 days of the outage completion. These inspection and evaluation reports are compiled biennially into a summary report [5]. Finally, Section 7.7 “Evaluation Requirement” requires that any engineering evaluation used to disposition relevant indications must be conducted in accordance with an NRC-approved methodology. WCAP-17096-NP-A is an example of such a methodology [6].

These requirements from MRP-227, Sections 7.5, 7.6, and 7.7 are all “Needed” elements under the NEI 03-08 protocol [7]. The protocol provides further governance of licensee response to emergent issues. In Appendix B of NEI 03-08, the following requirement is provided for emergent issues:

> *4. EMERGENT ISSUES*
>
> *Utilities shall inform the applicable IP* [issue program] *of significant emergent materials-related issues occurring at their plants when they have potential generic implications. In order to support this communication, each IP shall be prepared to perform a timely evaluation of the significance of emergent materials issues that fall within the scope of its program. The IP evaluation should be performed within a timeframe that supports the utility's needs where possible. Items that should be considered in the IP's evaluation include:*
>
> - *Safety significance*
> - *Demonstration of a new degradation type*
> - *Effect on the basis of industry guidance*
> - *Effect on the existing knowledge base*
> - *Expected regulatory significance*

Thus, observation of cracking degradation in a PWR core barrel weld would lead to notification of the appropriate industry issue program and transmittal of the inspection results to the EPRI program manager who would then transmit them to the NRC through the periodically updated summary report. The operating experience would also be entered into the licensee's corrective action program and require evaluation of the extent of condition and disposition through engineering evaluation.

Comparison of BWR and PWR Operating Experience and Environmental Conditions

The core shroud in the BWR design and the core barrel in the PWR design are very similar, in that they are large-diameter, thick-walled cylinders which are largely used to guide the coolant flow through the internals. They are even constructed of essentially the same materials: austenitic stainless steel plate rolled and welded into a cylinder. Although the function, fabrication, and materials of construction are similar for the two configurations, service experience at PWRs has been very different from that at BWRs.

A high-level discussion of the differences between the occurrence of SCC and IASCC in the two reactor designs was presented during the February 15, 2018 meeting between the NRC staff and industry representatives [1]. More detail supporting these discussions is provided in letter LTR-AMLR-18-18 [8], titled, “Comparison of Boiling Water Reactor and Pressurized Water Reactor Chemistry and Operating Experience with Austenitic Stainless Steel.” The goal of this white paper is to clearly identify the differences between the two designs, and to justify differences in the treatment of PWR core barrels as compared to BWR core shrouds.

The white paper provides technical arguments to identify the theoretical basis for the significant differences in cracking susceptibility, and it provides a summary review of the service experience with each reactor type to demonstrate the actual differences observed. These two approaches provide a complementary argument to support different treatments of these BWR and PWR components. The mechanical design criteria and temperatures of operation for the BWR core shroud and the PWR core barrel are similar, but there are major differences in the reactor coolant water chemistry to which they are exposed.

The primary difference is reflected in the electrochemical potential (ECP), which measures whether or not the environment is more oxidizing or more reducing. The PWR primary water environment is the most reducing case at approximately -770 mV_{SHE}. This is at least several hundred mV lower than BWR hydrogen water chemistry (HWC) environments and nearly 1000 mV lower than the ECP in a BWR normal water chemistry (NWC) regime. Operation below approximately -230 mV_{SHE} results in mitigation of SCC and IASCC. The beneficial effects of a reducing environment have been explained in Section 2 of [8]. The service experience supports these conclusions as well and has been reviewed in Sections 3 and 4 of [8].

BWR core shrouds, as well as other internals components fabricated from austenitic stainless steels, have experienced significant levels of SCC, while little SCC of austenitic stainless steels has been observed to date in PWR environments. Multiple detailed inspections under the requirements of MRP-227 have been completed on PWR core barrels, and only one recent inspection has discovered relevant indications. Since PWRs will operate for their entire licensed lifetimes with low ECP in the primary coolant system due to hydrogen additions, it is considered extremely unlikely that SCC or IASCC will occur in a PWR core barrel.

Limitations on Core Barrel Weld Accessibility and Reactor Vessel Internals Disassembly Risks

The accessibility of core barrel welds depends on the weld location and the design of the plant. Welds in the upper core barrel in both Combustion Engineering and Westinghouse-designed plants should have close to 100% accessibility; though some obstruction could occur due to gussets or other attachments or due to the proximity of the containment cavity to the core barrel while it is in the stand. Welds that are ground flush may also present inspectors with difficulties in reliably finding the weld location.

Welds located in the lower core barrel have significantly less accessibility. Inside of the core barrel, either the baffle-former assembly (Westinghouse) or core shroud assembly (Combustion Engineering) cover 100% of the inner diameter of the barrel in the beltline region. Below the beltline region, 100% of the inner diameter of the barrel is covered by the other lower internals.

For the outer diameter of the core barrel, the welds in the beltline region have several different levels of accessibility. Combustion Engineering-designed plants with no thermal shield approach 100% accessibility. Westinghouse-designed plants with thermal shields must access the welds through the gap between the core barrel and the thermal shield and will have less than 100% accessibility (the exact coverage will depend on the plant and the specific inspection tooling). Finally, Westinghouse-designed plants with neutron panels will have access to approximately 50-60% of the circumference. Axial welds that happen to be behind neutron panels could have significantly lower accessibility. Thermal shields and neutron panels do not cover the outer diameter of the core barrel below the beltline region, but the presence of attachments like radial keys and core snubber lugs could reduce the achievable coverage slightly below 100%.

One of the basic assumptions in MRP-227 development was that component disassembly should be avoided unless absolutely warranted. This was addressed under the response to request for additional information 4-8 for MRP-227-A [3]. Disassembly carries serious risks, including:

- Personnel safety and radiation exposure during disassembly, inspection, and re-assembly
- Operations to disassemble and reassemble would have to be performed remotely, which increases difficulty significantly and may lead to irreparable damage at any point in the operations
- Cutting, shaping, or removal operations can lead to loose parts and debris, which can impact fuel integrity
- Components with elevated irradiation dose cannot be welded, so extensive modification to the base internals component design may be required

The accessibility limitations and significant risks associated with component disassembly were considered by the industry when developing the core barrel weld inspection coverage requirements.

References:

1. Materials Reliability Program Presentation, "MRP-227, Revision 1 Requests for Additional Information," NRC Public Meeting, Rockville, MD, February 15, 2018. (ML18043B155)
2. Email from Joseph Holonich (NRC) to Kyle Amberge (EPRI), Subject: "Information We Want on the Docket from MRP-227, Rev. 1 RAI Discussion," February 22, 2018. (ML18053A058)
3. *Materials Reliability Program: Pressurized Water Reactor Internals Inspection and Evaluation Guidelines (MRP-227-A)*. EPRI, Palo Alto, CA: 2011. 1022863.
4. *Materials Reliability Program: Pressurized Water Reactor Internals Inspection and Evaluation Guidelines (MRP-227, Revision 1)*. EPRI, Palo Alta, CA: 2015. 3002005349.
5. *Materials Reliability Program: Inspection Data Survey Report* (MRP-219, Revision 11). EPRI, Palo Alto, CA: 2015. 3002005509.
6. Westinghouse Report, WCAP-17096-NP-A, Revision 2, "Reactor Internals Acceptance Criteria Methodology and Data Requirements," August 2016.
7. Guidelines for the Management of Materials Issues, NEI 03-08, Nuclear Energy Institute, Washington, DC, Latest Edition.
8. Westinghouse Letter, LTR-AMLR-18-18, Revision 2, "Comparison of Boiling Water Reactor and Pressurized Water Reactor Experience with Cracking of Austenitic Stainless Steel," April 24, 2018.

Responses to Supplemental Questions in Support of the Nuclear Regulatory Commission Safety Evaluation of MRP-227, Revision 1

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Framatome Inc. Report ANP-3610 Revision 1, dated 5/9/2018

Nature of Changes

Revision	Section(s) or Page(s)	Description and Justification
Rev. 000	All	Initial Issue
Rev. 001	All	Updated with new Framatome template
Rev. 001	All	Changed “AREVA” to “Framatome”
Rev. 001	viii	Added purpose of Rev. 001
Rev. 001	1-1	Added purpose of Rev. 001
Rev. 001	2-2 and 2-3	Clarified response to RAI 1 based on clarifications requested by NRC
Rev. 001	2-12	Clarified response to RAI 4 based on clarifications requested by NRC
Rev. 001	2-21	Updated Table 2-2 reference to “Note 12” and not “Note 10” and added “B2” to Primary Item assembly name
Rev. 001	2-23	Clarified response to RAI 17 based on clarifications requested by NRC
Rev. 001	2-29	Clarified response to RAI 21 based on clarifications requested by NRC
Rev. 001	3-1	Added references 5, 11, 12, 13, and 14 based on clarifications requested by NRC in Section 2 RAI responses and renumbered others appropriately

Contents

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Nomenclature

(If applicable)

Acronym	Definition
AMP	Aging Management Program
ANO-1	Arkansas Nuclear One Unit 1
ASME	American Society of Mechanical Engineers
B-B	Baffle-to-Baffle
B-F	Baffle-to-Former
B&PV	Boiler and Pressure Vessel
B&W	Babcock and Wilcox
CASS	Cast Austenitic Stainless Steel
CB-F	Core Barrel-to-Former
CMTR	Certified Material Test Report
CRGT	Control Rod Guide Tube
CSS	Core Support Shield
DB-1	Davis-Besse Unit 1
EPRI	Electric Power Research Institute
FMECA	Failure Mode, Effects and Criticality Analysis
IASCC	Irradiation-Assisted Stress Corrosion Cracking
IC	Irradiation-Enhanced Creep
IE	Irradiation Embrittlement
IMI	In-Core Monitoring Instrumentation
ISI	Inservice Inspection
ISR	Irradiation-Enhanced Stress Relaxation
MRP	Materials Reliability Program
NEI	Nuclear Energy Institute
NRC	Nuclear Regulatory Commission
OE	Operating Experience
PWR	Pressurized Water Reactor
RAI	Request for Additional Information
RV	Reactor Vessel
RVI[1]	Reactor Vessel Internals
RVLMS	Reactor Vessel Level Monitoring System
SCC	Stress Corrosion Cracking

[1] Used in quotations from the US NRC.

Acronym	Definition
S/N	Serial Number
SSHT	Surveillance Specimen Holder Tube
TE	Thermal Aging Embrittlement
US	United States
UT	Ultrasonic Testing
UTS	Upper Thermal Shield
VT-3	Visual Testing

ABSTRACT

By letter dated December 21, 2015, the Electric Power Research Institute (EPRI) submitted for U. S. Nuclear Regulatory Commission (US NRC) staff review Topical Report MRP-227, Revision 1, "Materials Reliability Program: Pressurized Water Reactor Internals Inspection and Evaluations Guidelines." The US NRC has issued Requests for Additional Information (RAIs) on this submittal. This report provides the Framatome Inc. responses for RAIs 1, 2, 3, 4, 6, 11, 13, 17, 18, 21, 22, 25, and 27. The primary purpose of Revision 001 of this report is to provide clarifications requested by the U.S. NRC regarding the responses to RAIs 1, 4, 17, and 21.

1.0 INTRODUCTION AND SUMMARY

By letter dated December 21, 2015, the Electric Power Research Institute (EPRI) submitted for U. S. Nuclear Regulatory Commission (US NRC) staff review Topical Report MRP-227, Revision 1, “Materials Reliability Program: Pressurized Water Reactor Internals Inspection and Evaluations Guidelines” (Reference 1). The US NRC has issued Requests for Additional Information (RAIs) on this submittal (Reference 2) and this report provides the responses to those RAIs assigned to Framatome Inc. The responses to these RAIs are supported by technical information contained in MRP-189, Revision 2 (Reference 3) and MRP-231, Revision 3 (Reference 4).

Upon receipt of the RAIs, EPRI and Framatome Inc. reviewed the RAIs and determined who would respond to each RAI. The responses for RAIs 1, 2, 3, 4, 6, 11, 17, 18, 21, 22, and 25 were assigned to Framatome Inc. The responses to RAIs 5, 7, 8, 9, 10, 12, 14, 15, 16, 19, 20, 23, 24, and 26 were assigned to Westinghouse by EPRI. The responses to RAIs 13 and 27 are shared by Framatome Inc. and Westinghouse. For completeness, this document contains sections for all twenty-seven RAIs; however, the responses for those RAIs that were assigned to Westinghouse only say “Assigned to Westinghouse by EPRI by the Division of Responsibility.”

The primary purpose of Revision 001 of this report is to provide clarifications to the responses to RAI 1 (Section 2.1.2), RAI 4 (Section 2.4.2), RAI 17 (Section 2.17.2), and RAI 21 (Section 2.21.2.2) based on requested clarifications from the U.S. NRC Staff. The clarifications requested by the US NRC Staff can be found in Reference 5.

2.0 REQUESTS FOR ADDITIONAL INFORMATION

The NRC RAIs are addressed, as noted in Section 1.0, in Section 2.1 through Section 2.27.

2.1 *RAI 1*

2.1.1 Statement of RAI 1

In MRP-227, Revision 1 (Reference 1), for the following Babcock & Wilcox (B&W) primary components, the schedule for the initial (baseline) examination changed from "during the next 10-year ISI [inservice inspection]" to "during the next 10-year ISI interval."

- B2. Control Rod Guide Tube (CRGT) Assembly - spacer castings
- B3. Vent Valve Assembly
  - A. Vent valve top retaining ring
  - B. Vent valve bottom retaining ring
- B10. Core Barrel Assembly-Baffle plates
- B11. Core Barrel Assembly - Locking devices, including locking welds, of baffle-to-Former bolts and internal baffle-to-baffle-bolts

Clarify what this means; for example, does "during the next 10-year ISI" mean during the next scheduled 10-year ISI examination of the reactor vessel internals (RVI), or does it mean sometime during the next 10-year ISI interval? If the latter, does that mean these examinations may not be performed until up to 20 years from now, if the current 10-year ISI interval started today? If this is the case, justify waiting up to 20 years to perform the baseline examination.

2.1.2 Response to RAI 1

The updated wording in MRP-227 Revision 1 (Reference 1) does not allow initial examinations 20 years into the period of extended operation. Per Section 4 on page 4-2 of MRP-227, Revision 1, the term "10-year ISI interval" is intended to mean the plant's existing schedule associated with removal of the core barrel, i.e., during the next scheduled 10-year ISI interval examination. As defined in the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel (B&PV) Code Section XI (Reference 6), the first inspection interval is defined as 10 years following the initial start of plant commercial startup, and successive inspection intervals are defined as 10 years following the previous inspection interval. Each inspection interval may be extended by as much as 1 year and may be reduced without restriction, provided the examinations required for the interval have been completed. Successive intervals shall not extend more than 1 year beyond the original pattern of 10 year intervals and shall not exceed 11 years in length. Therefore, for the baseline (i.e., initial) examinations, the intention of this wording is for examinations to be performed prior to the end of the fourth ASME ISI interval and not more than 11 years since the previous ASME ISI interval was completed, i.e., what is allowed by Section XI of the ASME B&PV Code. This is also consistent with the stipulations stated in Section 4.2.6 of MRP-227, Revision 1 for subsequent examination intervals.

The following update, to the MRP-227-Rev. 1 Table 4-1 "Examination Method/Frequency" column for the items related to this RAI (listed below), is proposed:

- B2. Control Rod Guide Tube (CRGT) Assembly - spacer castings
- B3. Vent Valve Assembly
  - A. Vent valve top retaining ring
  - B. Vent valve bottom retaining ring
- B10. Core Barrel Assembly-Baffle plates

- B11. Core Barrel Assembly - Locking devices, including locking welds, of baffle-to-former bolts and internal baffle-to-baffle-bolts

Current wording:

Visual (VT-3) examination during the next 10-year ISI interval.

Subsequent examination during each 10-year ISI interval.

Proposed wording:

Visual (VT-3) examination during the next 10-year ISI interval.

Subsequent examination during each 10-year ISI interval.

(Note 10)

Add Note 10 to Table 4-1 as follows:

10. The term “10-year ISI interval” is intended to mean the plant’s existing schedule associated with removal of the core barrel, i.e., during the next scheduled 10-year ISI interval examination. As defined in the ASME Boiler and Pressure Vessel (B&PV) Code Section XI, the first inspection interval is defined as 10 years following the initial start of plant commercial startup, and successive inspection intervals are defined as 10 years following the previous inspection interval. Each inspection interval may be extended by as much as 1 year and may be reduced without restriction, provided the examinations required for the interval have been completed. Successive intervals shall not extend more than 1 year beyond the original pattern of 10 year intervals and shall not exceed 11 years in length.

2.2 *RAI 2*

2.2.1 Statement of RAI 2

For MRP-227-A item in Table 4-1, "B&W Plants Primary Components:"

<u>Plenum Cover Assembly & Core Support Shield Assembly</u>

Plenum cover weldment rib pads

Plenum cover support flange

Core Support Shield (CSS) top flange

The revised item description in MRP-227, Revision 1, Table 4-1 is:

"B1.Plenum Cover Assembly & Core Support Shield Assembly"

a. Plenum cover weldment rib pads

b. Plenum cover support flange

c. Plenum cover support ring

d. CSS top flange"

The examination coverage changed from "Determination of differential height of top of plenum rib pads to reactor vessel seating surface, with plenum in reactor vessel" to "Determination of differential height of top of plenum rib pads/plenum cover support ring location to reactor vessel seating surface, with plenum in reactor vessel."

The change to this item was to add the plenum cover support ring as a subcomponent and to add this subcomponent as an additional reference point for the physical measurement.

The plenum cover support ring appears to be a new subcomponent added in MRP-227, Revision 1. The plenum cover support ring is addressed in MRP-189, Revision 1, “Screening, Categorization, and Ranking of B&W-Designed PWR Internals Component Items” (Reference 6[2]), and was determined to be Category A. The plenum cover assembly – weldment rib pads and plenum cover assembly – support flange were determined to be Category C for wear in MRP-189. Therefore, the NRC staff requests that the Electric Power Research Institute (EPRI) clarify why the plenum cover support ring was added as a subcomponent and how and why the support ring was added as a reference location for making the physical measurements.

2.2.2 Response to RAI 2

During preparation of MRP-189, Revision 2 (Reference 3), the preparers determined that the plenum cover support ring, in addition to the plenum rib pads, is in contact with the reactor vessel closure head and that additional wording clarification should be included. Core clamping measurements have been performed consistently at all B&W-designed units and have included the plenum cover support ring. Therefore, the text in MRP-189, Revision 2 has been updated to reflect this change.

[2] “Reference 6” is a direct quotation from the NRC RAI letter (Reference 2). The same reference herein is Reference 8.

The plenum cover support ring and the plenum rib pads are machined to a common plane (see Figure 2-3 and accompanying text in Section 2.3.1 of MRP-189, Revision 2). Figure 2-1 (Reference 7) provides a sketch of the profile view of the core clamping details. Figure 2-2 (Reference 7) provides a sketch showing the interference height measurement. In both figures, the solid vertical blue box depicts the plenum cover support ring and the horizontal white box depicts a plenum rib pad. In Figure 2-1 only, the horizontal patterned box depicts the plenum cover support flange. The core clamping area is composed of three mating surfaces: the location where the core support shield upper flange sits on the reactor vessel ledge, the location where the plenum cover support flange sits on the upper core support shield flange, and (of interest to this RAI) the area where the common plane of the plenum rib pads and top of the plenum cover support ring interface with the reactor vessel closure head. The actual clamping occurs when the reactor vessel closure head is positioned on the reactor vessel and the studs are tensioned. The stud loading is principally resisted by the interface between the reactor vessel and the reactor vessel closure head. The force transmitted through the internals between the reactor vessel closure head flange and the reactor vessel flange is referred to as the core clamping force. Therefore, in addition to MRP-189, both MRP-231 and MRP-227 were updated accordingly.

Figure 2-1. Sketch Showing Core Clamping Components to be Measured (Reference 7)

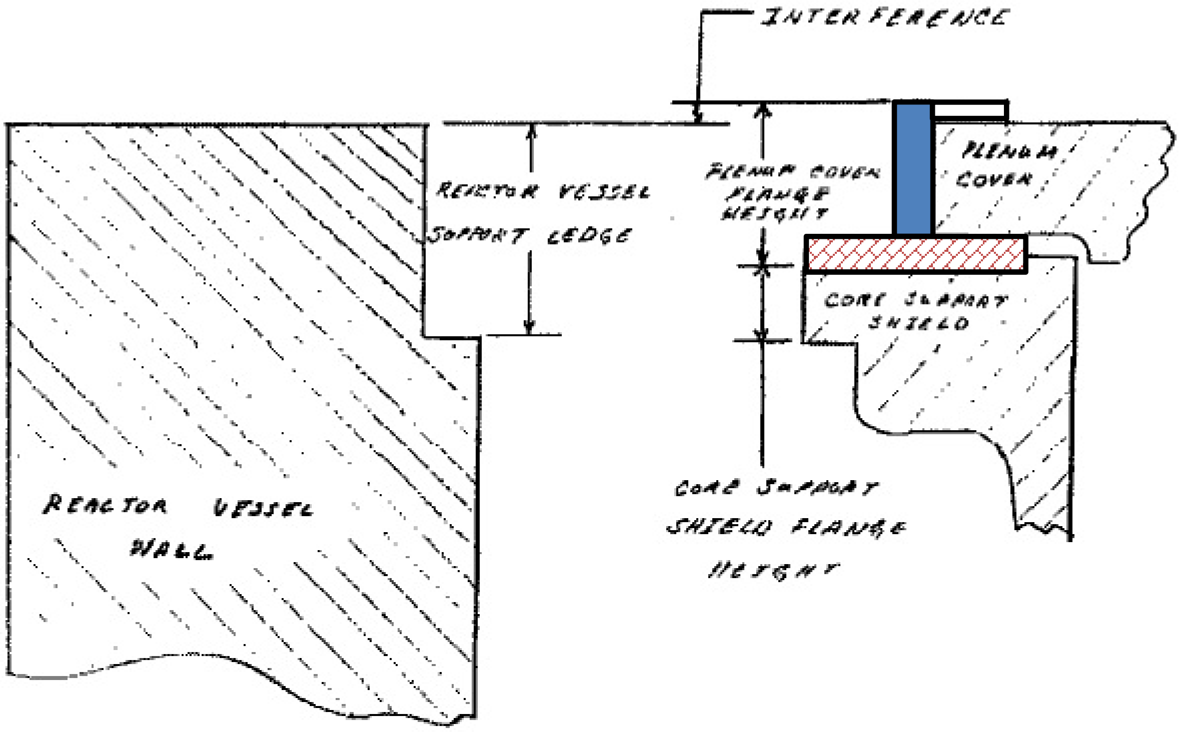
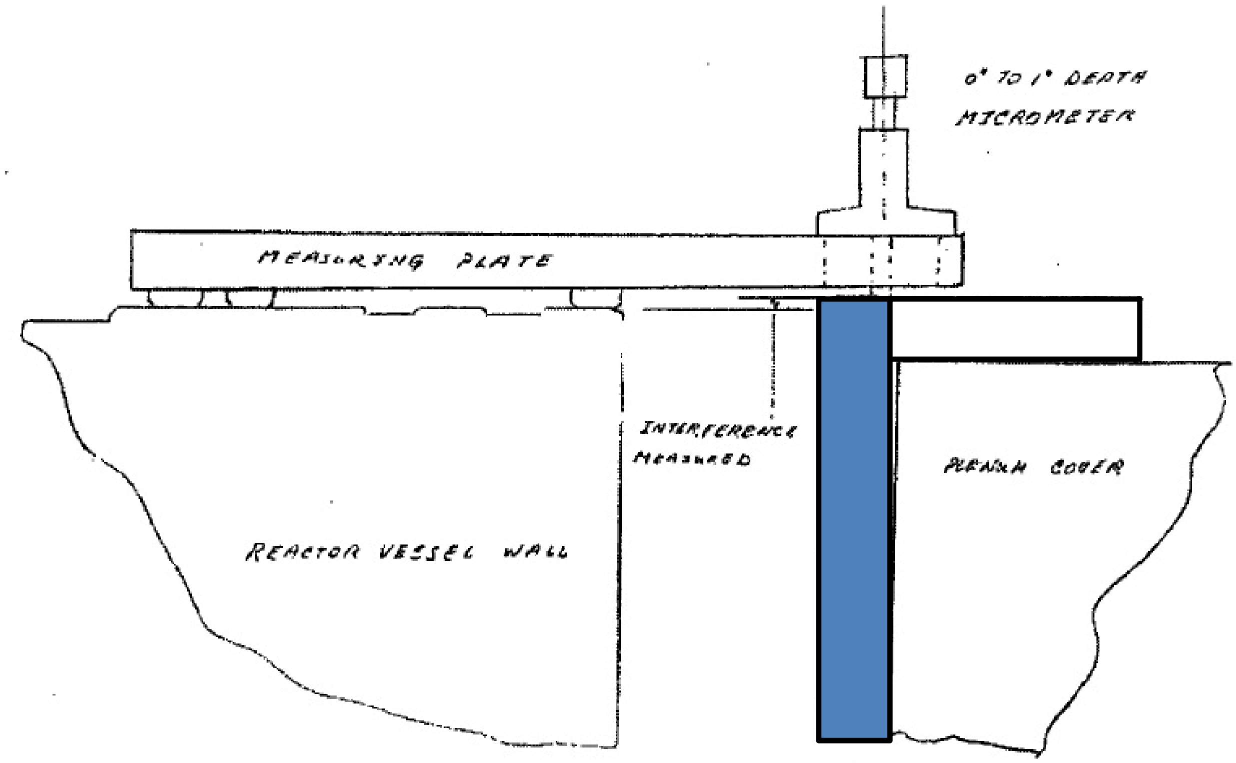


Figure 2-2. Sketch Showing Interference Height Measurement (Reference 7)



2.3 *RAI 3*

2.3.1 Statement of RAI 3

In Table 4-1, the "Control Rod Guide Tube Assembly – CRGT," spacer castings previously had no expansion link. An expansion link to the vent valve bodies has now been added in MRP-227, Revision 1. The vent valve bodies were not an expansion component in MRP-227-A. According to MRP-189, Revision 1, the vent valve bodies are cast austenitic stainless steel (CASS), as are the CRGT spacer castings. Since the vent valve bodies were previously a no additional measures component, the U.S. Nuclear Regulatory Commission (NRC) staff requests that EPRI explain why the vent valve bodies were made an expansion component for the CRGT spacer castings.

2.3.2 Response to RAI 3

In the evaluation performed for originally installed vent valve bodies, all of the vent valve bodies were previously screened out (i.e., Category A for all aging degradation mechanisms of concern) in MRP-189, Revision 1 (Reference 8) due to their low ferrite content. However, as noted in Section 3.2, item J.2 (page 3-7) and in Table 3-2 Note 6 (page 3-15) of MRP-189, Revision 2 (Reference 3), each Babcock and Wilcox (B&W) utility was also supplied with spare vent valves in addition to the originally installed vent valves. It is now known that some spare vent valves have been installed, replacing the original vent valves. Currently, the certified material test reports (CMTRs) for spare vent valves are not readily available. Therefore, because the ferrite content of these replacement vent valve bodies is not currently known, the vent valve bodies are currently assumed to have a ferrite content exceeding 20%.

Per Section 3.2.6.2 of MRP-231, Revision 3 (Reference 4), the CRGT spacer castings are made of Type CF3M castings. The vent valve bodies are made of Type CF8 castings. The vent valve bodies and the CRGT spacer castings are located above the core and their operating conditions are similar, i.e., at hot leg temperature with an irradiation dose too low to cause irradiation embrittlement (IE). Hence, their extent of thermal aging embrittlement (TE) is expected to be similar. However, Type CF3M material contains 2% to 3% percent molybdenum, which may potentially contribute to a higher TE for the CRGT spacer castings than the Type CF8 vent valve bodies, depending on the casting method and ferrite content. Therefore, the CRGT spacer castings are categorized as Primary component items, because the potential degradation is expected to bound that of the vent valve bodies. The vent valve bodies are thus categorized as Expansion component items for the CRGT spacer castings.

Additionally, as remarked in Note 4 in Table 4-4 of MRP-227, Revision 1 (Reference 1), each utility may be able to identify the body serial number (S/N) and heat number for each installed vent valve during vent valve exercising. If the CMTR and chemical composition of the vent valve body are subsequently found and hence the ferrite content of the currently installed vent valve bodies can be verified to be under the TE screening criterion for the material, the vent valve bodies can be removed as an Expansion component item and default back to a No Additional Measures component item. As an alternative, each utility may be able to utilize a statistical argument for estimating the ferrite content of the currently installed vent valve bodies using the results obtained in PWROG-15032-NP (Reference 9). If, at any time, the ferrite content of any installed vent valve body is concluded to be greater than the screening criterion for TE in MRP-175 (Reference 10), that vent valve body will need to be included as an Expansion component item.

2.4 *RAI 4*

2.4.1 Statement of RAI 4

In Table 4-1, the schedule for the initial (baseline) ultrasonic (UT) examination of the Core Barrel Assembly - Baffle-to-former Bolts changed from “no later than two refueling outages from the beginning of the license renewal period” to “volumetric (UT) examination during the next 10-year ISI interval.” Since it is not clear when the next ten-year ISI interval starts (it could be up to ten years from the current date), this could result in the baseline examination being significantly later than MRP-227-A would require. It was not clear to the NRC staff whether this assumes all six operating B&W units have completed baseline UT examinations already. The NRC staff, therefore, requests EPRI provide the following information:

1. Does the initial baseline UT examination schedule for the baffle-to-former bolts in MRP-227, Revision 1 assume an examination of baffle-to-former bolts has been completed within two refueling outages from the beginning of the period of extended operation?

2. If not, justify the change in the schedule for the initial (baseline) UT examination of the baffle-to-former bolts.

2.4.2 Response to RAI 4

2.4.2.1 Part 1

Yes, the initial baseline UT examination schedule for the baffle-to-former (B-F) bolts in MRP-227, Revision 1 (Reference 1) assumes an examination of B-F bolts has been completed within two refueling outages from the beginning of the period of extended operation for each operating B&W unit.

As of January 2018, four of the six B&W units have performed the baseline UT examinations of the B-F bolts (see NRC Accession Numbers ML14135A383-ML14135A385 (Reference 11) and ML16144A789 (Reference 12) for details of these UT examinations). The two remaining units are considering ceasing operations before their UT examination commitment dates (see NRC Accession Numbers ML17171A151 (Reference 13) and ML18094A834 (Reference 14)).

The following update, to the MRP-227-Rev. 1 Table 4-1 "Examination Method/Frequency" for the baffle-to-former bolts, is proposed:

Current wording:

> *Volumetric (UT) examination during the next 10-year ISI interval.*
>
> *Subsequent examination during each 10-year interval.*

Proposed wording:

> *Volumetric (UT) examination during the next 10-year ISI interval.*
>
> *Subsequent examination during each 10-year interval.*
>
> *(Note 11)*

Add Note 11 to Table 4-1 as follows:

> *11. This assumes that all units operating as of December 2011 have performed baseline (initial) volumetric (UT) examinations no later than two refueling outages from the beginning of their first license renewal period.*

2.4.2.2 Part 2

A response to this part of this RAI is not required to be provided based on the response to Part 1 of this RAI.

2.5 *RAI 5*

Assigned to Westinghouse by EPRI by the Division of Responsibility

2.6 *RAI 6*

2.6.1 Statement of RAI 6

In Table 4-1, Item B11., "Core Barrel Assembly – Locking Devices," including locking welds, of baffle-to-former bolts and internal baffle-to-baffle bolts, has applicable aging mechanisms of irradiation assisted stress corrosion cracking (SCC) (IASCC), irradiation embrittlement (IE) including the detection of missing, non-functional, or removed locking devices or welds, and has as an Expansion link "locking devices, including locking welds, of the external baffle-to-baffle bolts and core barrel-to-former bolts." However, in MRP-227, Revision 1, a new Note 8 has been added for the expansion link, which states that "the aging degradation mechanism of IASCC is only applicable to the baffle-to-former bolt and internal baffle-to-baffle bolt locking devices, not the baffle-to-former bolt and internal baffle-to-baffle bolt locking device welds." However, under the expansion link column in Table 4-1, the expansion link for Item B11 is described as locking devices, including locking welds, of the external baffle-to-baffle bolts and core barrel-to-former bolts.

The NRC staff therefore requests the following information:

a) Clarify whether the expansion link column or Note 8 is correct.

b) If Note 8 is correct, explain why IASCC is not applicable to the locking device welds, and why there are no Expansion links for the welds.

2.6.2 Response to RAI 6

2.6.2.1 Part a

Note 8 in Table 4-1 and the Expansion link in MRP-227, Revision 1 (Reference 1) are both correct.

As discussed in Section 3.2.3 and stated in Note 1 of Table 3-3 in MRP-231, Revision 3 (Reference 4), the locking devices for the baffle-to-former (B-F) bolts and internal baffle-to-baffle (B-B) bolts are Primary component items for irradiated-assisted stress corrosion cracking (IASCC) with no Expansion link. Therefore, Note 8 of Table 4-1 in MRP-227, Revision 1 is correct as written. However, for clarification, the “Effect (Mechanism)” column of Table 4-1 for Item B11., “Core Barrel Assembly—Locking devices, including locking welds, of baffle-to-former bolts and internal baffle-to-baffle bolts” in MRP-227, Revision 1 will be revised as follows, with the reference notes remaining the same as those currently in MRP-227, Revision 1 Table 4-1:

Table 2-1. New Effect (Mechanism) Entry for Primary Core Barrel Assembly Locking Devices and Locking Welds in Table 4-1 of MRP-227, Revision 1

Primary Item	Effect (Mechanism)
B11.Core Barrel Assembly Locking devices, including locking welds, of baffle-to-former bolts and internal baffle-to-baffle bolts	Locking Devices: Cracking (IASCC, IE) including the detection of missing, non-functional, or removed locking devices (Note 3) Locking Welds: Cracking (IE) including the detection of missing, non-functional, or removed locking device welds (Note 3, 8)

2.6.2.2 Part b

As stated in Note 1 of Table 3-1 in MRP-189, Revision 2 (Reference 3), page B-7 of MRP-175 (Reference 10) suggests an IASCC fluence screening criterion of 2E21 n/cm^2, E > 1.0 MeV (3 dpa) for highly stressed component items such as bolts, springs, and multi-pass welds. As shown in Table 3-3 of MRP-189, Revision 2, the welds for the B-F bolt locking devices to the baffle plates (WC-15, WC-94), the welds for the internal B-B bolt locking devices to the baffle plates (WC-16, WC-17), the welds for the CB-F bolt locking devices to the core barrel cylinder (WC-14, WC-15), and the welds for the external B-B bolt locking devices to the baffle plates (WC-16, WC-134, WC-135) are either fillet, locking, or plug welds, not multi-pass welds. Therefore, the aging degradation mechanism of IASCC is screened out (i.e., Category A) for these locking device welds and there are no Primary/Expansion relationships for IASCC. As noted in Table 2-1, these welds are considered potentially susceptible to irradiation embrittlement (IE).

2.7 *RAI 7*

Assigned to Westinghouse by EPRI by the Division of Responsibility

2.8 *RAI 8*

Assigned to Westinghouse by EPRI by the Division of Responsibility

2.9 *RAI 9*

Assigned to Westinghouse by EPRI by the Division of Responsibility

2.10 *RAI 10*

Assigned to Westinghouse by EPRI by the Division of Responsibility

2.11 *RAI 11*

2.11.1 Statement of RAI 11

In Table 4-4, B&W Plants Expansion Components, Item B7.1 Core Barrel Assembly – Upper Thermal Shield (UTS) bolts and their locking devices and Item B7.2 Core Barrel Assembly - Surveillance specimen holder tube (SSHT) bolts and their locking devices, had changes to the “Effect (mechanism)” information. Specifically, irradiation creep/irradiation stress relaxation (IC/ISR)/Wear/Fatigue were added for the SSHT bolts.

Note 7 has also been added to Table 4-4 indicating that this table entry for the SSHT bolts also includes the aging degradation mechanisms of ISR/IC/wear/fatigue for the compression collar and washer for the SSHT bolt. The compression collars for the SSHT bolt are not included in the screening and failure mode, effects and criticality analysis (FMECA) documented in MRP-189, "Materials Reliability Program: Screening, Categorization, and Ranking of B&W-Designed PWR Internals Component Items” (Reference 6[3]) and MRP-190, "Materials Reliability Program: Failure Modes, Effects, and Criticality Analysis of B&W-Designed PWR Internals” (Reference 7[4]).

The NRC staff, therefore, requests EPRI:

a) Explain why the new aging mechanisms of IC/ISR/Wear/Fatigue was added for the SSHT bolts.

b) Clarify whether the compression collars were left out of the screening and FMECA process as an oversight, or whether the compression collars are the same as the SSHT bolt locking cups and tie plates that are included in the screening and FMECA. If the latter, explain why the screening and FMECA results for these components changed.

[3] “Reference 6” is a direct quotation from the NRC RAI letter (Reference 2). The same reference herein is Reference 8.
[4] “Reference 7” is a direct quotation from the NRC RAI letter (Reference 2). The same reference herein is Reference 15.

2.11.2 Response to RAI 11

2.11.2.1 Part a

As noted in Table 3-1 in MRP-189, Revision 2 (Reference 3), the screening criteria for irradiation-enhanced stress relaxation and creep (ISR/IC) is a dose of greater than or equal to 1.3E20 n/cm^2, E > 1 MeV (0.2 dpa), applied to all bolted or spring locations. Additionally, wear and fatigue are applicable to bolted or spring items where ISR/IC is applicable. Table 3-2 in MRP-189, Revision 2 lists the dose at 54 EFPY for the surveillance specimen holder tube (SSHT) bolts as 1.5E21 n/cm^2, E > 1 MeV (note that the 48 EFPY fluence value listed in MRP-189, Revision 1 (Reference 8) is 7.60E19 n/cm^2, E > 1 MeV). Therefore, since the fluence value at 54 EFPY for the SSHT bolts exceeds the ISR/IC screening criteria, ISR/IC, wear, and fatigue are applicable to the SSHT bolts. As noted in Table 4-2 of MRP-189, Revision 2, the failure modes, effects, and criticality analysis (FMECA) categorized the susceptibility to ISR/IC/wear/fatigue of the SSHT bolts as Category “C” and the safety consequence as “1”, with a final categorization as Category “B” (note; this categorization accounts for stress corrosion cracking [SCC] in addition to ISR/IC/wear/fatigue). Therefore, the age-related degradation mechanisms of ISR/IC/ wear/fatigue of the SSHT bolts are included in MRP-227, Revision 1.

2.11.2.2 Part b

The only currently-operating B&W unit with functional SSHT bolts is Davis-Besse Unit 1 (DB-1). The original SSHT bolts were replaced at DB-1 in 1984 and 1990; a records search was completed in 2010, after the publication of MRP-189, Revision 1 (Reference 8) and MRP-190 (Reference 15), which identified that the design of these SSHT bolts at DB-1 utilizes a bolt, compression collar, spherical washer, and a tie-plate/crimp locking cup assembly. These additional components items, which were not known to exist during preparation of MRP-227-A, are now listed in Table 3-2 of MRP-189, Revision 2 (Reference 3) and screen as Category Not A for various aging degradation mechanisms, including ISR/IC/wear/fatigue for the compression collar. The revised FMECA documented in Table 4-2 of MRP-189, Revision 2 determined the compression collar to be Category B due to ISR/IC/wear/fatigue. As discussed in MRP-231, Revision 3 (Reference 4), Section 3.2.4, the SSHT bolts, including the compression collars, are Expansion component items in MRP-227, Revision 1 (Reference 1).

2.12 *RAI 12*

Assigned to Westinghouse by EPRI by the Division of Responsibility

2.13 *RAI 13*

2.13.1 Statement of RAI 13

The following four areas pertaining to Tables 4-1 through 4-3 could be informed by Operating Experience (OE) related to the actual accessibility achieved for primary system components during baseline inspections:

1. In MRP-227, Revision 1, has OE been used to modify or clarify examination coverage requirements of MRP-227-A based on the actual accessibility achieved during the examinations completed to date? If so, identify the components that have had examination coverage revised based on OE, and describe the reason for the change. If coverage requirements have not been revised based on OE, justify why this has not been done.

2. Has OE with actual coverage achieved resulted in any primary component that was previously considered to be accessible being reclassified as inaccessible, either because of the percentage of the component surface area, length, or population that is accessible was insufficient to provide reasonable assurance of functionality, or because insufficient coverage was achieved of the most likely portion of the component to exhibit degradation? Identify any primary components that have been reclassified as inaccessible and identify what alternate measures, such as an engineering analysis, were taken to provide reasonable assurance of component functionality.

3. For primary components reclassified as inaccessible, were the expansion links reevaluated for these components?

4. For any primary components reclassified as inaccessible, were alternate primary components selected?

2.13.2 Response to RAI 13

2.13.2.1 Part 1

Operating experience (OE) has not yet been used to modify or clarify examination coverage of MRP-227-A based on actual accessibility achieved during the examinations completed to date for the Babcock and Wilcox (B&W)-designed units. Coverage has not yet been revised based on OE because, except for one instance, the expected coverage (including allowances for minimum populations of bolts and locking devices) has been able to be achieved during each examination to date. At one US PWR, only about 99% coverage was achieved during the control rod guide tube (CRGT) spacer casting VT-3 examination; 10 spacer castings at each of the 4 screw locations were inaccessible for VT-3 examination due to permanent obstruction from the reactor vessel level monitoring system (RVLMS) installed at one CRGT location. This OE was not available during preparation of MRP-227, Revision 1 (Reference 1). As a result, the examination coverage in MRP-227, Revision 1 for this Table 4-1 entry will be modified as noted by underlined text in Table 2-2:

Table 2-2. New CRGT Spacer Casting Entry for Table 4-1 of MRP-227, Revision 1

Primary Item	Applicability	Effect (Mechanism)	Expansion Link (Note 1)	Examination Method/Frequency (Note 1, 2)	Examination Coverage
B2. Control Rod Guide Assembly CRGT spacer castings	All plants	Cracking (TE), including the detection of fractured spacers or missing screws (Note 3)	Vent valve bodies	Visual (VT-3) examination during the next 10-year ISI interval. Subsequent examination prior to the end of each 10-year ISI interval.	Accessible surfaces at each of the four screw locations (at every 90º) of 100% of the CRGT spacer castings. (limited accessibility) (Note 12) See Figure 4-5.

Notes to Table 2-2:

1. Examination acceptance criteria and expansion criteria for the B&W component items and welds are detailed in Table 5-1.

2. Initial examinations may be scheduled concurrently with the next 10-year ISI but shall be performed no later than the final outage of the current 10-year ISI interval at the time of entry into the period of extended operation, except where otherwise noted.

3. Loss of ductility and fracture toughness, which are induced by thermal aging or neutron irradiation embrittlement, are not directly monitored. These effects are indirectly managed by performing periodic visual examinations capable of detecting cracking in the component.

12. At one of the CRGT locations at ANO-1, a device was installed to monitor water level that required removal of the center control rod drive mechanism and installation of an assembly that blocks access to all of the CRGT spacer castings except the top spacer casting. Therefore, for all plants except ANO-1, 100% of the CRGT spacer castings refers to 690 CRGT spacer castings. There is no requirement to examine this set of CRGT spacer castings at this one CGRT assembly at ANO-1 and 100% is defined as 680 CRGT spacer castings for ANO-1.

2.13.2.2 Part 2

Operating experience (OE) to-date has not resulted in the reclassification of a component as inaccessible for the Babcock and Wilcox (B&W)-designed units.

2.13.2.3 Part 3

A response to this part is not required based on the response to Part 2 of this RAI.

2.13.2.4 Part 4

A response to this part is not required based on the response to Part 2 of this RAI.

2.14 *RAI 14*

Assigned to Westinghouse by EPRI by the Division of Responsibility

2.15 *RAI 15*

Assigned to Westinghouse by EPRI by the Division of Responsibility

2.16 *RAI 16*

Assigned to Westinghouse by EPRI by the Division of Responsibility

2.17 *RAI 17*

2.17.1 Statement of RAI 17

For Table 4-4, B&W Plants Expansion Items, Core Barrel Assembly, B11.1.Locking Devices, including locking welds, of the external baffle-to-baffle bolts and core barrel-to-former bolts, the primary link changed from:

"…locking devices, including locking welds, of baffle-to-former bolts **or** internal baffle-to-baffle bolts," to

"B11.Locking devices, including locking welds, of baffle-to-former bolts **and** internal baffle-to-baffle bolts.

Does the change from “or” to “and” mean degradation now has to be exhibited in both the locking devices for the baffle-to-former bolts and the locking devices for the internal baffle-to-baffle bolts for the expansion to be required, whereas in MRP-227-A the expansion would be required if only one of these items exhibited degradation? If so, justify the changes.

2.17.2 Response to RAI 17

No, the change from “or” to “and” noted in this RAI does not mean degradation now has to be exhibited in both the locking devices for the baffle-to-former bolts and the locking devices for the internal baffle-to-baffle bolts for the Expansion to be required. The Primary link for the B11.1 Expansion items was, and still is, required to be both types of locking devices. This change was editorial in nature only. The same text is now consistently used in Tables 4-1, 4-4, and 5-1 of MRP-227, Revision 1 (Reference 1).

The following update, to Note 1 of the MRP-227-Rev. 1 Table 4-4, is proposed:

Current wording:

> *Examination acceptance criteria and expansion criteria for the B&W component items and welds are in Sections 5.1 - 5.3 of this document.*

Proposed wording:

> *Examination acceptance criteria and expansion criteria for the B&W component items and welds are in Sections 5.1 - 5.3 of this document. Degradation exhibited in any of these primary links requires expansion in accordance with the criteria noted in Table 5-1.*

2.18 *RAI 18*

2.18.1 Statement of RAI 18

In Table 4-4, the Lower Grid Assembly – Item B10.3., “Lower Grid Rib Section,” has been added as an additional Expansion link for Primary Item B10., “Core Barrel Assembly – Baffle Plates. Lower Grid Assembly,” – Item B10.3., “Lower Grid Rib Section,” was not included in MRP-227-A as either a Primary or Expansion component. The NRC staff therefore requests EPRI explain why this item has apparently been recategorized from “no additional measures” to “expansion.”

2.18.2 Response to RAI 18

The following text is summarized in Table 2-3.

The lower grid rib section was screened as Category Not A for irradiation embrittlement (IE) in Table 3-2 of MRP-189, Revision 1 (Reference 8) and MRP-189, Revision 2 (Reference 3). The lower grid rib section was categorized as Category “A” as a part of the FMECA in MRP-189, Revision 1, owing the categorization to a susceptibility ranking of “B” and a safety consequence of “1”.

As detailed in Note 14 on page 4-13 of MRP-189, Revision 2, the safety consequence of failure of the lower grid rib section was increased from that assigned in MRP-190 (Reference 15) (safety consequence of “1” to a safety consequence of “2”) based on its direct core support function (the need to assign a safety consequence of “2”) and redundancy because of the need for multiple flaws to initiate and grow to critical flaw size in multiple ribs to cause a safety concern (the need to not assign a safety consequence of “3”). Additionally, MRP-189, Revision 2 introduced a new table (Table 4-1), not in MRP-189, Revision 1, detailing the susceptibility metrics for IE, irradiation-enhanced stress relaxation and creep (ISR/IC), and irradiation-assisted stress corrosion cracking (IASCC). Based on Table 4-1 of MRP-189, Revision 2, the aging degradation susceptibility metric of the lower grid rib section to IE is a “C” due to the expected 54 EFPY dose of 3.2E21 n/cm^2, E > 1 MeV listed in Table 3-2 of MRP-189, Revision 2. Therefore, the preliminary categorization was revised to Category “C” based on a susceptibility ranking of “C” and a safety consequence of “2”.

Section 3.2.3.1 of MRP-231, Revision 3 (Reference 4) describes the classification of the lower grid rib section as an Expansion item based on its expected dose as compared to the baffle plates; this is reflected in Table 3-1 of MRP-227, Revision 1 (Reference 1).

Table 2-3. Summary of Changes for the Lower Grid Rib Section between MRP-227-A and MRP-227, Revision 1

	MRP-189 Screening Result	MRP-189 FMECA Result	MRP-231 Aging Management Strategy
Lower Grid Rib Section in MRP-227-A	Category "Not A" for IE	Category "A" based on a susceptibility of "B" and safety consequence of "1"	Category "A" (not considered in MRP-231)
Lower Grid Rib Section in MRP-227, Revision 1	Category "Not A" for IE	Category "C" based on a susceptibility of "C" and safety consequence of "2"	Expansion (to Baffle Plates)

2.19 *RAI 19*

Assigned to Westinghouse by EPRI by the Division of Responsibility

2.20 *RAI 20*

Assigned to Westinghouse by EPRI by the Division of Responsibility

2.21 *RAI 21*

2.21.1 Statement of RAI 21

In Table 5-1, the Core Barrel Assembly – Baffle-to-former bolts expansion criteria have changed. In MRP-227-A, the Expansion criteria is “Confirmed unacceptable indication in greater than or equal to 5% (or 43) of the baffle-to-former bolts, provided that none of the unacceptable bolts are on former elevations 3, 4, and 5, or greater than 25% of the bolts on a single baffle plate, shall require an evaluation of the internal baffle-to-baffle bolts for the purpose of determining whether to examine or replace the internal baffle-to-baffle bolts. The evaluation may include external baffle-to-baffle bolts and core barrel-to-former bolts for the purpose of determining whether to replace them.” In MRP-227, Revision 1, the expansion criteria is “Confirmed unacceptable indications in greater than or equal to 5% of the baffle-to-former bolts (including previously failed/removed bolts) shall require an evaluation of the baffle-to-baffle bolts and the core barrel-to-former bolts by the completion of the next refueling outage. The evaluation shall also assess functionality of the core barrel assembly with aging degradation of the baffle-to-baffle bolts and core barrel-to-former bolts for the purpose of determining continued operation or replacement.”

The criteria requiring expansion if greater than 25 percent of the bolts on one baffle plate are degraded would result in expansion if clustering of degraded bolts was present, which has been seen in recent OE with baffle-former bolt degradation in Westinghouse-design RVI. It is also not clear why the language regarding bolts on former elevations 3, 4, and 5 has been removed from the expansion criteria.

The NRC staff therefore requests that EPRI provide the technical basis for the changes to the expansion criteria for the baffle-to-former bolts in B&W plants. The response should address the following items:

a) An explanation for the removal of the language from the expansion criteria related to bolts on former levels 3, 4, and 5, and whether this results in less conservativism. If less conservative, provide a justification for the reduction in conservatism.

b) Why was the expansion criterion of more than 25% of the bolts on a single plate [degraded] removed in Revision 1 especially considering recent OE with clustered baffle-former bolt degradation?

2.21.2 Response to RAI 21

2.21.2.1 Part a

The Expansion criteria were updated to only include consideration that an active age-related degradation mechanism in the baffle-to-former (B-F) bolts would be present, as aging degradation drives the Expansion inspections. A structural analysis performed for the operating Babcock and Wilcox (B&W) units demonstrated that failure of B-F bolts on all rows other than rows 3, 4, and 5 was an acceptable configuration for continued operation, as that configuration maintained fuel impact assessment boundary conditions. Additional details regarding this structural analysis are in Sections 3.2.2.1.3 and 3.2.2.1.5 of MRP-231, Revision 3 (Reference 4). Since the presence of these B-F bolts is a consideration for continued operation, and not Expansion, associated language was removed from the Expansion criteria related to the B-F bolts. As this language is not associated with potential core barrel-to-former (CB-F) bolt, internal baffle-to-baffle (B-B) bolt, or external B-B bolt failures as predicted by B-F bolt failures, removal of the language does not result in less conservatism.

2.21.2.2 Part b

As noted above in the response to Part a of this RAI, the Expansion criteria were updated to only include consideration that an active age-related degradation mechanism in the baffle-to-former (B-F) bolts would be present, as aging degradation drives the Expansion inspections. The exact mechanism(s) of irradiation-assisted stress corrosion cracking (IASCC) is not fully understood but the current consensus is that IASCC results from the synergistic effects of irradiation damage to the material, the aggressive water environment, and the stress state. The water environment is similar for all B&W B-F bolts. The accumulated dose at 40-60 years may not exceed the industry-accepted IASCC screening criteria of 3 dpa (Reference 10) for some of these austenitic stainless steel B-F bolts, but there still may be some irradiation-enhanced stress relaxation and creep (ISR/IC), thus decreasing the stresses for these B-F bolts. For B-F bolts with very high dose, ISR/IC could decrease the stress such that little stress is remaining in the B-F bolt. Therefore, irradiation damage and stress state are potential competing interests for IASCC initiation.

Removal of the 25% criteria, which may be an operability concern, on one baffle plate is appropriate because Framatome has evaluated for the B&W design that failures of B-F bolts on a given baffle plate have a negligible impact on the likelihood of failure of the associated CB-F bolts.

The compilation of all recent operating experience (OE) with clustered B-F degradation being observed at some Westinghouse-designed units was not yet available during the preparation of MRP-227, Revision 1. However, this OE has been assessed and it was determined that several key stress drivers for IASCC appear to exist at the Westinghouse-designed units that do not exist at the B&W-designed units (Reference 16). These are: 1) higher applied stresses due to the pressure differential inherent in the downflow configuration at the applicable Westinghouse-designed units (B&W units have always been an upflow configuration), and 2) bolting installation and design characteristics that appear to be more susceptible to cracking (i.e., bolt length, initial bolt torque levels, the fabrication process associated with some internal hex head designs, and potentially higher stress concentration factors in the head-to-shank area). While sufficient information is currently not available to Framatome Inc. to completely quantify these stress drivers, less IASCC initiation is expected in the B&W units. This is supported by the current OE for the B&W units (one B-F bolt with a UT indication in over 3,400 B-F bolts examined), which has a high likelihood of being a random failure and not an indication of an active degradation mechanism having initiated (Reference 16).

A cluster of failed B-F bolts is thought to be the result of a random distribution of degraded B-F bolts (by IASCC) near to each other where the degraded B-F bolts lose their ability to carry their expected load and consequently nearby or neighboring B-F bolts carry increased loads, leading to adjacent failures (Reference 17), which is more likely to occur with a downflow configuration.

The requirement for greater than 25% of the B-F bolts on a single baffle plate to require Expansion to the CB-F, internal B-B, and external B-B bolts would indicate that clustering of IASCC initiation would be expected. This type of clustering is not expected with the B&W units since higher applied stresses due to failure of neighboring B-F bolts from a downflow configuration are not inherent to the B&W design due to the lower pressure differential; therefore, such an Expansion criterion is not necessary.

2.22 *RAI 22*

2.22.1 Statement of RAI 22

In Table 5-1, the examination acceptance criteria and expansion criteria for the Core Barrel Assembly – Baffle Plates have changed. In MRP-227-A, the examination acceptance criteria column in Table 5-1 stated that the specific relevant condition is readily detectable cracking in the baffle plates. In MRP-227, Revision 1, this has been changed to state the specific relevant condition is readily detectable cracking connecting openings in the baffle plates (i.e., each bolt hole and flow hole).

With respect to expansion criteria, in MRP-227-A, the expansion criteria states;

> Confirmed cracking in multiple (2 or more) locations in the baffle plates shall require expansion, with continued operation of former plates and the core barrel cylinder justified by evaluation or by replacement by the completion of the next refueling outage.

In MRP-227, Revision 1, the expansion criteria state:

> Gross cracking (if confirmed) within one inch of a bolt or flow hole location in the baffle plates shall require:
>
> a) An evaluation of the former plates and the core barrel cylinder for the purpose of determining continued operation or repair/replacement by the completion of the next refueling outage. Alternatively, repair/replacement activities may be initiated based on results of a best effort former plate and core barrel cylinder examination.
>
> b) That the Visual (VT-3) examination be expanded by the completion of the next refueling outage to include 100% of the accessible portions of the lower grid rib section heat-affected zones adjacent to the in-core monitoring instrumentation (IMI) guide tube spider-to-lower grid rib section welds.

The relevant condition now requires cracking connecting openings in baffle plates, rather than just detectable cracking. Also, the expansion criteria in MRP-227, Revision 1 seem inconsistent with the relevant conditions since the relevant conditions require linkage of openings by cracking, while the expansion criteria only seem to require cracking within one inch of an opening.

The NRC staff therefore requests the following information:

a) Provide a technical justification for the change in the definition of the relevant condition for the baffle plates, specifically, the new requirement that the cracking link openings in the baffle plates.

b) Provide a technical justification for the change in the expansion criteria for the baffle plates.

c) Clarify whether expansion is only required if cracking links two or more openings or whether expansion would be required if cracking is present within one inch of any opening.

2.22.2 Response to RAI 22

2.22.2.1 Part a

The relevant condition from Table 5-1 of MRP-227-A will be retained in MRP-227, Revision 1 (Reference 1). This is reflected by changes (strikethrough of text) shown in the “Primary Item Examination Acceptance Criteria” column in Table 2-4 in this document.

2.22.2.2 Part b

The Expansion criteria from Table 5-1 of MRP-227, Revision 1 (Reference 1) were updated for two reasons: 1) previously, MRP-227-A required two instances of confirmed cracking before Expansion; MRP-227, Revision 1 only requires one instance of confirmed gross cracking (within one inch of a bolt or flow hole, which is the examination coverage for both MRP-227-A and MRP-227, Revision 1), and 2) the Expansion criteria were updated to reflect Licensee/Applicant Action Item 6 from MRP-227-A (part a of the Expansion criteria) and the lower grid rib section as an Expansion item to the baffle plates (see response to RAI 18 in Section 2.18.2 of this document, part b of the Expansion criteria).

Table 2-4. New Baffle Plate Entry for Table 5-1 of MRP-227, Revision 1

Primary Item	Applicability	Primary Item Examination Acceptance Criteria (Notes 1,2)	Expansion Link(s)	Expansion Criteria	Expansion Item Examination Acceptance Criteria
Core Barrel Assembly Baffle Plates	All plants	Visual (VT-3) examination The specific relevant condition is readily detectable cracking ~~connecting openings~~ in the baffle plates ~~(i.e., each bolt hole and flow hole).~~	Former plates Core barrel cylinder (including vertical and circumferential seam welds) Lower grid rib section	Gross cracking (if confirmed) within one inch of a bolt or flow hole location in the baffle plate shall require: a) An evaluation of the former plates and the core barrel cylinder for the purpose of determining continued operation or repair/replacement by the completion of the next refueling outage. Alternatively, repair/replacement activities may be initiated based on results of a best effort former plate and core barrel cylinder examination. b) That the VT-3 examination be expanded by the completion of the next refueling outage to include 100% of the accessible portions of the lower grid rib section heat-affected zones adjacent to the IMI guide tube spider-to-lower grid rib section welds.	N/A

Notes to Table 2-4:

1. The examination acceptance criterion for visual examination is the absence of the specific relevant condition(s).

2. Refer to MRP-231 (Reference 4) Section 3.3 for additional details on examination acceptance and expansion criteria.

2.22.2.3 Part c

As stated in Table 2-4, expansion is required if there is confirmed gross cracking within one inch of a bolt or flow hole location in the baffle plate.

2.23 *RAI 23*

Assigned to Westinghouse by EPRI by the Division of Responsibility

2.24 *RAI 24*

Assigned to Westinghouse by EPRI by the Division of Responsibility

2.25 *RAI 25*

2.25.1 Statement of RAI 25

In Table 4-1, or Item B15. “IMI Guide Tube Assembly Spiders and Spider welds,” – the examination coverage changed from “100% of top surfaces of 52 spider castings and welds to the adjacent lower grid rib section” in MRP-227-A to “Spiders: 100% of the accessible top surfaces and 100% of the accessible spider surfaces adjacent to the spider casting welds” and “Spider welds: 100% of the accessible welds to the adjacent lower grid rib section.”

The NRC staff requests that EPRI explain why the description of the examination coverage for this item changed, and explain the significance of this change.

2.25.2 Response to RAI 25

The description of the examination coverage was updated to clarify the examination coverage by separating the incore monitoring instrumentation (IMI) guide tube spiders and the welds to identify the specific areas of concern. The addition of the words “accessible” throughout the examination coverage description for these items have been added to maintain consistency with the examination coverage descriptions of other components items and welds in MRP-227, Revision 1 (Reference 1).

2.26 *RAI 26*

Assigned to Westinghouse by EPRI by the Division of Responsibility

2.27 *RAI 27*

2.27.1 Statement of RAI 27

The MRP-227, Revision 1 report includes Section 7.3, "Reactor Internals Guidelines Inspection Requirement." In this section, EPRI states "[e]ach commercial U.S. PWR unit shall implement the requirements of Tables 4-1 through 4-9 and Tables 5-1 through 5-3 for the applicable design."

Section 7.3 in MRP-227, Revision 1, omitted the following information that was previously included in Section 7.3 of the MRP-227-A report:

> Consistent with the requirements of Nuclear Energy Institute (NEI) document NEI 03-08 (Reference 18), if the guidance contained in Table 4-1 through 4-9 and/or Tables 5-1 through 5-3 cannot, need not, or will not be implemented as written, a technical justification must be prepared that clearly states what requirement cannot, need not, or will not be met and why; what alternative action is being taken to satisfy the objective or intent of the guidance; and, why the alternative action is acceptable. Examples of alternatives that may be justifiable are: elevation of an Expansion component to Primary; substitution of an equivalent or more rigorous examination than is required by the tables; or destructive testing in lieu of nondestructive examination, such as the case where one or more of the primary components is being replaced. Since the Expansion components are also "needed" requirements, the technical justification for not fully implementing a Primary component examination or not implementing it in a manner consistent with its intent, would be expected to include disposition of the associated Expansion components.

When submittal of a deviation from work products or elements is required, the justification shall be reviewed and approved in accordance with the applicable plant procedures with the additional responsibility for deviation from a 'Needed' element that an internal independent review is performed and that concurrence is obtained from the responsible utility executive. Further, as stipulated in the Implementation Protocol (Appendix B) of NEI 03-08, a utility is required to notify the Issue Program (e.g., the EPRI Materials Reliability Program, EPRI-MRP) and the NRC.

Justify the basis for omitting these paragraphs from the scope of Section 7.3 of the MRP-227, Revision 1 report.

2.27.2 Response to RAI 27

This text was omitted because key parts of the previous text, such as the requirement to provide a technical justification for a deviation from a Needed or Mandatory Requirement or the contents of the NEI 03-08 (Reference 18) Implementation Protocol, are included by reference to NEI 03-08.

3.0 REFERENCES

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2. Letter, Joseph J. Holonich, (NRC Senior Project Manager) to Brian Burgos, (MRP Program Manager), “Request for Additional Information for Electric Power Research Institute Topical Report MRP-227, Revision 1, ‘Materials Reliability Program: Pressurized Water Reactor Internals Inspection and Evaluations Guideline’,” (CAC NO. MF7740), May 15, 2017, NRC Accession Number ML17079A027.

3. Materials Reliability Program: Screening, Categorization, and Ranking of Babcock & Wilcox-Designed Pressurized Water Reactor Internals Component Items and Welds (MRP-189, Revision 2). EPRI, Palo Alto, CA: 2014. 3002004238.

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Comparison of Boiling Water Reactor and Pressurized Water Reactor Experience with Cracking of Austenitic Stainless Steel

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PURPOSE

This Enclosure transmits a comparison of boiling water reactors (BWRs) and pressurized water reactors (PWRs). The focus of this comparison is the experience with stress corrosion cracking (SCC) of austenitic stainless steel alloys in the primary coolant of each reactor design and the fundamental chemistry differences that contribute to those differences. This comparison supports the current Nuclear Regulatory Commission (NRC) safety evaluation of the MRP-227, Revision 1 PWR Reactor Vessel Internals Inspection and Evaluation Guidelines.

Ref:
Westinghouse Electric Co. LTR-AMLR-18-18, Revision 2, dated 4/24/2018

Comparison of Boiling Water Reactor and Pressurized Water Reactor Experience with Cracking of Austenitic Stainless Steel

1 Introduction and Background

The core shroud in the boiling water reactor (BWR) design and the core barrel in the pressurized water reactor (PWR) design are very similar, in that they are large-diameter, thick-walled cylinders which are largely used to guide the coolant flow through the internals. They are even constructed of essentially the same materials, austenitic stainless steel plate, rolled and welded into a cylinder. Although the function, fabrication, and materials of construction are similar for the two configurations, service experience at PWRs has been very different from that at BWRs. Nonetheless, from a regulatory viewpoint, it could appear to be expedient, albeit technically inaccurate, to apply the more extensive degradation experience with BWRs to PWR components. The goal of this white paper is to clearly identify the differences between the two designs, and to argue that these two components should be treated differently.

Technical arguments will be provided to identify the theoretical basis for the significant differences in cracking susceptibility. Additionally, the service experience in PWRs and BWRs will be reviewed to demonstrate the actual differences observed. These two approaches provide a complementary argument to support different treatments of these BWR and PWR components.

2 Theoretical Basis for the Observed Difference in Degradation between BWRs and PWRs

The large differences between BWRs and PWRs in environmental cracking operating experience clearly indicate some underlying difference in the parameters affecting degradation. The observed cracking in the austenitic stainless steel components of BWRs has been due to either stress corrosion cracking (SCC) or, in the case of irradiated components, irradiation-assisted SCC (IASCC) [1]. PWR austenitic stainless steel components have also experienced SCC and IASCC, but to a much lesser extent [1] [2].

Comparison of the SCC and IASCC behavior in these two reactor designs requires evaluation of the parameters relevant to these cracking mechanisms. Any type of SCC requires three conditions to be met at the same time: 1) a susceptible material 2) subjected to adequate tensile stress 3) in an aggressive enough environment (See Figure 2-1 [3]). Most of these can be dispositioned as being equivalent between the BWR and PWR designs, since both types are light water reactors operating in similar environmental regimes. The biggest difference is the hydrogen overpressure, which scavenges effectively all the oxygen from the PWR water, and virtually eliminates the possibility of SCC in that environment.

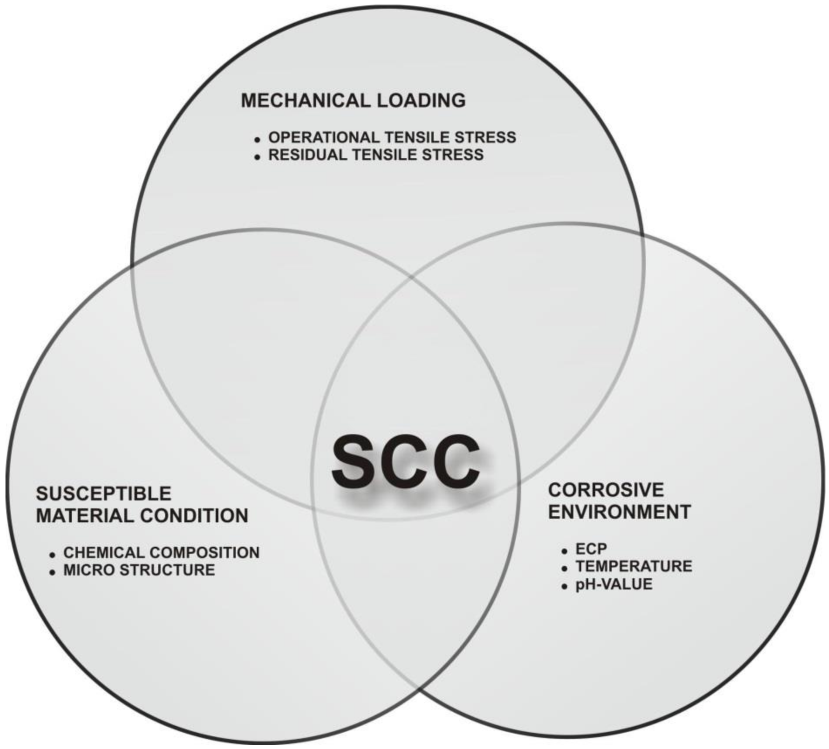

Figure 2-1: Synergistic effects which together lead to SCC

Comparison of Parameters Affecting SCC and IASCC

The three pre-requisites for SCC are shown in Figure 2-1:

- Mechanical loading (tensile)
- Susceptible material
- Corrosive environment

Each of these areas summarizes multiple possible conditions that could impact the occurrence of SCC in a PWR or BWR. Some of these are listed on the circles in Figure 2-1, but those listed on the figure are not exhaustive.

Mechanical Loading

Mechanical loading must be tensile to create SCC or IASCC. Four types of stress are typically present in a PWR or BWR:

- Gravity stresses – due to deadweight loads
- Design mechanical stresses – due to bolt preload, spring forces
- Operational stresses – due to fluid flow, temperature differences, gamma heating
- Residual stresses – due to welding, grinding, or cold working

From a very high level, the basic designs of PWRs and BWRs are the same. Energy is extracted from neutron irradiation using water and transformed into electricity by making steam. The temperature ranges and pressures are similar enough that the same types of materials and similar overall shape and layout of the reactors can be used. Austenitic stainless steel materials are used for most of the reactor vessel internals components of both designs [1] [4] [5], so the same limits on mechanical design parameters are applicable.

The similarities in the two design types mean that both PWRs and BWRs have components subjected to significant deadweight loads and both have components subjected to either bolt preload or spring forces. Operationally, both design types have significant fluid flow, temperature differences, and gamma heating. The gamma heating in certain components of a PWR will actually be higher than that in a BWR due to the higher radiation dose rate on internals components like baffle plates and former plates [3] [4] [6]. Finally, both designs contain the potential for residual stress in large structural welds or components which may have experienced surface grinding or cold work during fabrication.

Based on these potential sources of mechanical loading, it can be concluded that the BWR and PWR designs are essentially equivalent in the potential to have tensile stresses that lead to SCC or IASCC. The higher dose rate in certain localized areas of the PWR may lead to those components having slightly higher susceptibility. Overall, there are plenty of locations in each reactor design with enough tensile stress to support environmental cracking. This has been borne out in the operating experience to date [1] [2] [7, Appendix A], and discussed in Sections 3 and 4 of this report.

Susceptible Material

The focus of this white paper is on evaluating the differences between the environmental cracking experienced in the austenitic stainless steel components of the BWR and PWR reactor vessel internals. Austenitic stainless steel makes up the majority of the components in both reactor designs. Most of the components are fabricated from Type 304, with Types 316 and 347 making up nearly all of the rest [4] [5]. The materials specified for both reactor design types are governed by Section II of the ASME Boiler and Pressure Vessel Code [8]. Thus, large differences in material composition, heat treatment, and mechanical properties are not expected.

Cold work has a strong detrimental effect on the resistance to SCC of austenitic stainless steels in both BWRs and PWRs [2] [9]. Bulk cold work has been limited by the requirement to keep yield strength below 90 ksi under Code Case N-60-6 [10] and Regulatory Guide 1.84 [11]. All nuclear equipment vendors have processes in place to limit the cold work present in operating plant components. Such limitations are standard practice, and the ASME Boiler and Pressure Vessel Code also contains requirements to limit the amount of cold work in components. The presence of cold work in PWR reactor vessel internals was also summarized in report PWROG-15105-NP [12].

ASME Section III Division I Subsection NH [13] provides the following guidance regarding cold forming of reactor components subsequent to solution anneal and thermal treatment and is applicable to the fabrication of PWR core barrels and BWR core shrouds:

NH-4212 Effects of Forming and Bending Processes

The rules of this paragraph shall supplement those of NB-4212 and NB-4213. Any process may be used to form or bend pressure retaining materials, including weld metal, provided that the requirements of the subparagraphs below are met.

(a) Post fabrication heat treatment [in accordance with (b) below] of materials which have been formed during fabrication, shall be required unless one of the following conditions are met.

(1) <u>Maximum fabrication induced local strains[10] do not exceed 5%[11], regardless of the service temperature.</u>

(2) Written technical justification shall be provided in the Design Report for not performing heat treatment, subsequent to straining, or for the use of an alternate heat treatment procedure, to that specified in (b) below for fabrication induced strains greater than 5%.

Notes:

10. *Strain is defined as the maximum local fiber elongation or contraction per unit length; and where more than one strain increment occurs (e.g., biaxiality or reversed bending), it shall be the sum of the absolute values of all the strain increments.*
11. *Strain resulting from final straightening operations performed on materials furnished in the solution annealed or heat treated condition need not be included in the computation of strain.*

The commentary in note 11 above regarding straightening would apply to the as-received plate to fabricate the core barrel, and not to the final as-welded component. As such, manufacturers would need to limit maximum bending strain to 5%. The strain requirement should be calculated based on the actual post-weld distortion values, but is not expected to approach this 5% limit.

In addition, material specifications that govern the use of stainless steel materials generally refer to ASME B&PV Code Section II material specifications with the requirement that material is to be furnished in the solution-annealed condition. These specifications also typically require supplemental thermal treatments for base metal materials with cold work.

Cold straightening of base metal materials is generally prohibited. Cold bending operations performed on these materials are required to be followed by a stress relieving heat treatment. Repairs performed on these materials are generally followed by surface conditioning to remove residual surface tensile stresses resulting from the repair. It is also expected that field fit-up and auxiliary processes that could introduce cold work would only introduce minor increases, with the most significant increases in cold work being local and limited to the material surface [12].

Limiting cold work based on these requirements substantially reduces the risk for cold work-induced SCC [3]. This only leaves potential fabrication-induced cold work as a possible concern. This would include such things as surface repairs and temporary supports welded to the surface and removed after transport. There is evidence that repairs and temporary supports are both applicable to the BWR reactor vessel internals, where alignment lugs and transport "spiders" were welded to the core shroud for use during transport and installation [14]. Similar records have not been found for PWR reactor vessel internals; however, the lack of cracking events in PWRs has not provided motivation for an exhaustive search of fabrication records. Logically, one would expect that supports may have been required during the transport of large PWR internals components like the core barrel and that surface repairs involving grinding were almost certainly used.

Sensitization of austenitic stainless steel materials used in BWRs was a cause for early cracking observed in components exposed to reactor coolant [1]. Sensitized materials are also expected in the older PWR plant components, but experience to date has shown that the low oxygen environment of PWR primary coolant has not caused SCC due to sensitization [1].

Based on the similarity in the materials used and the general design and fabrication similarities, it can be concluded that there are no general material trends that explain the differences in BWR and PWR material degradation.

Corrosive Environment

The general lack of significant differences between BWR and PWR designs when considering stress and material leaves the corrosive environment in the reactor coolant as the likely culprit for the more aggressive environmental cracking behavior observed in BWRs. The effects of coolant chemistry have been studied at length for both reactors, and several parameters govern the impact of SCC or IASCC:

- Temperature
- Presence of impurities
- Electrochemical potential and the presence of hydrogen or oxygen
- pH and the presence of buffering agents
- Neutron irradiation dose (IASCC only)

The propensity for SCC and IASCC are directly related to the temperature, increasing with higher temperature and decreasing with lower temperature [1] [15]. This effect is the same for both PWRs and BWRs. When evaluating the potential for crack growth, a representative temperature of 288°C (550°F) can be used for a BWR, while a representative temperature of 325°C (617°F) can be used for PWR primary water [15]. In reality, the temperature varies throughout the coolant system in both cases, but this difference illustrates that PWR reactor vessel internals components are generally exposed to higher temperature in the hottest parts of the system than BWR components. This temperature difference results in a calculated factor of 5.6 higher crack growth rate for PWR conditions in the hottest locations over BWR conditions with hydrogen additions [15].

The reactor coolant in both BWRs and PWRs is well-controlled for potential impurities. The BWRVIP BWR water chemistry guidelines govern the purity of BWR coolant [16] and the EPRI PWR Primary

Water Guidelines govern the purity of PWR coolant [17]. Both of these guidelines control the presence of cation impurities like chloride and sulfate. The BWR guidance also has control parameters for coolant conductivity and guidance on other parameters that should be measured, such as dissolved metals, electrochemical potential (ECP), oxygen, and hydrogen. The PWR guidance includes additional control parameters for the measurement of fluoride, lithium, hydrogen, and oxygen. The guidelines provide a detailed technical basis for the selection of each of the required control parameters, which are typically aimed at reducing radiation fields or avoiding environmental cracking and corrosion. Earlier plant operation may not have targeted or achieved the tight control levels for impurities specified in the modern guidance, as shown for earlier BWR operation [18]. During prior operation, BWRs and PWRs could have also experienced off-normal chemistry for brief periods due to unexpected conditions like resin ingress. Modern management of the water chemistry has trended toward significantly better control, as shown for BWRs in [18]. Management according to the guidelines reduces the overall likelihood of SCC and IASCC due to impurities.

The BWR and PWR water chemistry guidelines [16] [17] provide guidance for the presence of dissolved oxygen and hydrogen in the reactor coolant. Oxygen is an oxidizing agent and hydrogen acts as a reducing agent. Practically, this means that metals will be more likely to form oxides and hydroxides in an aqueous environment with excess oxygen, for example Fe_2O_3 or $Fe(OH)_2$. In an environment with excess hydrogen, the less oxidized compounds and the base metals will be stabilized. The driving force for oxidation or reduction can be measured directly through the ECP of the system. A more positive ECP indicates a more oxidizing environment, and a more negative ECP indicates more reduction. In a typical BWR coolant environment, the threshold for SCC or IASCC is approximately -230 millivolts (mV) versus a standard hydrogen electrode (SHE) [1] [16] [19]. PWRs operate with between 25 and 50 cc of H_2/kg of H_2O (at standard temperature and pressure [STP]) [17], which keeps the ECP near approximately -770 mV [20]. This typical ECP in PWR primary water is well below the threshold level and far below the ECP of a BWR running with normal water chemistry. To address the issues associated with the high ECP of normal water chemistry (NWC), BWRs have gradually moved into using other chemistry management approaches. There are three broad BWR chemistry management approaches: NWC, hydrogen water chemistry (HWC), and noble metal chemistry.

The first type of chemistry operation in BWRs was NWC, and HWC and noble metal chemistry were added as options later to address specific issues [16]. Within the past decade, NWC was still used at some plants [21]; though all of the U.S. domestic BWRs had converted to HWC or later methods of ECP control [22]. Under NWC operation, a BWR uses high-purity water with no solid, liquid, or gas purposely added to the coolant [16]. Neutron radiation causes radiolysis of the water, creating oxidizing species that raise the ECP of NWC coolant to approximately +200 to +250 mV_{SHE} [23]. Issues with SCC led to the introduction of HWC at many plants. The addition of 0.3 to 2.0 ppm of hydrogen to the feedwater achieves SCC mitigation with some margin past the ECP threshold value, typically reducing the ECP by 500 mV_{SHE} [23]. The downside of HWC is that it causes a significant increase in the main steam line radiation and it also requires significantly more hydrogen addition than one would calculate based on what is needed at the metal surface. These issues led to the creation of noble metal chemistry [16]. Noble metal chemistry increases the efficacy of the hydrogen additions by treating the coolant system surfaces with a catalyst that lowers the activation energy for the hydrogen to oxygen

recombination reactions. This lowers the amount of hydrogen addition required and as a bonus lowers the radiation activity in the main steam line of a BWR.

EPRI published an extensive review of IASCC crack growth in [15]. This report showed that IASCC crack growth rates in austenitic stainless steel materials under light water reactor conditions can be divided into two categories. BWR NWC conditions made up one category and exhibited higher crack growth rates. BWR HWC and PWR chemistry conditions were lumped together in the other category and together exhibited lower crack growth rates. Similar conclusions are expected to apply to both the initiation of cracking and SCC without irradiation effects.

BWRs and PWRs manage the pH of the reactor coolant quite differently. BWR coolant is high-purity water with no deliberate additions to control pH [16], because the boiling action would result in a lack of chemistry control within the reactor core. PWR coolant is high-purity water with additions of boron as boric acid and lithium as lithium hydroxide [17]. The boron is used as part of the core reactivity control and lithium is added to maintain the pH neutrality of the coolant [1]. The pH of BWR coolant at temperature is approximately 5.65 and that of PWR coolant is controlled to around 7.2 [24]. Testing has been performed at multiple boron and lithium concentrations for PWR coolant resulting in the finding that at low ECP there is little impact from these changes on SCC crack growth [24]. Some recent testing has found that high lithium levels (at least by 8 ppm Li and greater) can lead to increased IASCC initiation rates [25]. This level of Li is higher than that typically used by PWRs, but determination of the effect of lithium concentrations between 2 and 8 ppm has not been completed.

Neutron irradiation dose has an impact on the potential for IASCC degradation. IASCC initiation testing in PWR primary water conditions has shown a strong effect of increasing irradiation dose in decreasing the stress required for cracking when at low doses (< 10 displacements per atom [dpa]) followed by a smaller effect at higher doses [3]. PWR components near the core experience much higher radiation doses than similar BWR components by approximately a factor of 10 [3]. Thus, from a radiation damage standpoint, the PWR components are expected to be more impacted by IASCC. However, components in both reactor types are subject to the strong dose dependence below 10 dpa, and the PWR components that experience radiation beyond 10 dpa do not experience the same strong increase in susceptibility with the additional dose.

Distinguishing which corrosive environment is more aggressive based on the foregoing discussions is difficult because of the presence of competing effects. The PWR environment has higher temperature and dose, which would lead to a higher propensity for SCC or IASCC. A BWR running with NWC conditions would have a higher likelihood of SCC and IASCC due to the excess oxygen. A BWR running with HWC implemented has been shown to have an IASCC crack growth rate behavior similar to that seen with PWR chemistry and about 4.6 times less than that in NWC [15]. Previous results have shown this to be as high as a factor of 10, while for sensitized unirradiated material the benefit of a lower ECP can be a factor of 20 to 50 [22]. Constant extension rate tensile testing of samples fabricated from the same heat of Type 304 austenitic stainless steel in both BWR NWC and PWR primary water conditions showed that the NWC caused IASCC but the PWR environment did not [20]. Impurity contents within the allowable values from the BWR or PWR water chemistry guidelines should not cause higher cracking risk, but the water purity during past operation is not well-known. The other parameters,

including factors affecting tensile stresses, pH and buffering agents, and material variables, are not expected to differentiate between BWRs and PWRs.

However, the evidence points to the strong impact of the oxidizing environment used in BWRs operating with NWC. This operating regime has a stronger impact on the likelihood of SCC or IASCC than do the higher temperature and dose experienced by materials in PWRs. PWRs have used low ECP for the entirety of their operation, whereas BWRs have operated for varying lengths of time with ECP levels above the thresholds for IASCC or SCC. These periods with higher ECP would have allowed the initiation and growth of cracks in the BWRs. PWRs have not experienced such cracking.

3 BWR Cracking Experience Summary

Intergranular SCC is a commonly observed phenomenon in BWR core shrouds. It was first observed in 1990 at the Mühleberg site in Switzerland [26]. As of 2013, the cracks being monitored in the core shroud of Mühleberg ranged from one inch to over two feet in length and had depths approaching one inch. In response to this, the original equipment manufacturer (OEM) issued an information letter to all owners of BWRs designed by the OEM, recommending a visual examination of the shroud circumferential welds [27]. The NRC responded to the early observations of cracking at several plants by issuing Generic Letter 94-03 to request that all licensees inspect their BWR core shrouds no later than the next scheduled refueling outage [28]. By the mid-1990s, the cracking had progressed to the point that several plants had installed tie rods to bolster the integrity of the core shroud, in the presence of circumferential cracks [26] [27]. By the late 1990s, a significant percentage of operating BWRs had installed tie rods [14] [29].

The tie rods relied on the integrity of the vertical welds of the shroud to maintain their key purpose, but, by 1993 cracking was being observed in the vertical welds as well [29]. Multiple visual and UT exams were conducted after this observation, and these cracks have been observed to progress slowly. Inspection and evaluation guidelines for the BWR core shroud were developed and documented in BWRVIP-76-A [30], which was submitted to the NRC for review, and Revision 1 received a safety evaluation in 2014 [31]. This supported establishment of up to a ten year re-inspection interval for these welds. To date, plants have been able to continue operation with the observed flaws despite increases in the number and size of the flaws.

As these inspections progressed, the number of flaws observed has increased and flaw dimensions have changed. Figure 3-1 illustrates the various types of cracking that have been observed [14]. Cracks oriented along the welds have been commonly observed and were the primary concern related to core shroud integrity when SCC was first identified in the early 1990s. More recently, “off-axis” cracking (i.e., cracks that propagate approximately perpendicular to the associated weld, see Figure 3-1) has become more of a concern as inspections have appeared to reveal changes to the off-axis cracks along with new or deeper than expected cracks. These changes observed during inspections could be due to improvements in inspection techniques and equipment rather than additional crack growth or initiation [14]. Analyses of the BWR core shroud indicate that the off-axis flaws are likely to grow through-wall but are not likely to grow into long cracks due to the stress distribution predicted around the weld [32]. These analytical results are consistent with the majority of the reported off-axis flaws.

Correlation studies have shown that off-axis cracking is most strongly related to plant design, neutron fluence, and shroud fabricator [14]. The location of most of the reported off-axis cracks has been the core shroud beltline region spanning the beltline welds H3 and H6a. Factors introduced during construction such as cold work due to manipulation of the shroud during fit-up or local material changes due to construction supports and alignment lugs that were welded to the core shroud and then removed could have contributed to the cracking. The correlation with neutron fluence is driven by the location of the cracking in beltline region welds (Figure 3-2) and the occurrence of the cracking in older units with more exposure time (Figure 3-3). Some of these correlated factors could be confounded with fabricator or design differences. The cause of this off-axis cracking remains under active investigation, but analytical

results indicate that off-axis cracks should remain short and do not have a significant impact on core shroud structural integrity [32]. To date, this has been confirmed by the results of field inspections.

Many other BWR internals components have also shown evidence of environmental cracking; though the cracking in core shrouds is one of the most widespread [26] and is the most relevant in a comparison to the PWR core barrel. The experience summarized above shows that stainless steel welds operating in BWR environments are rather highly susceptible to stress corrosion cracking. Neutron irradiation appears to increase this susceptibility. The experience for the BWR core shrouds does show evidence that much of the cracking initiated early in BWR plant life due to fabrication factors and the initial exposure to the oxidizing NWC environment. As will be seen in Section 4 below, the same materials operating in PWR environments have not displayed the same level of susceptibility and did not experience the early life degradation.

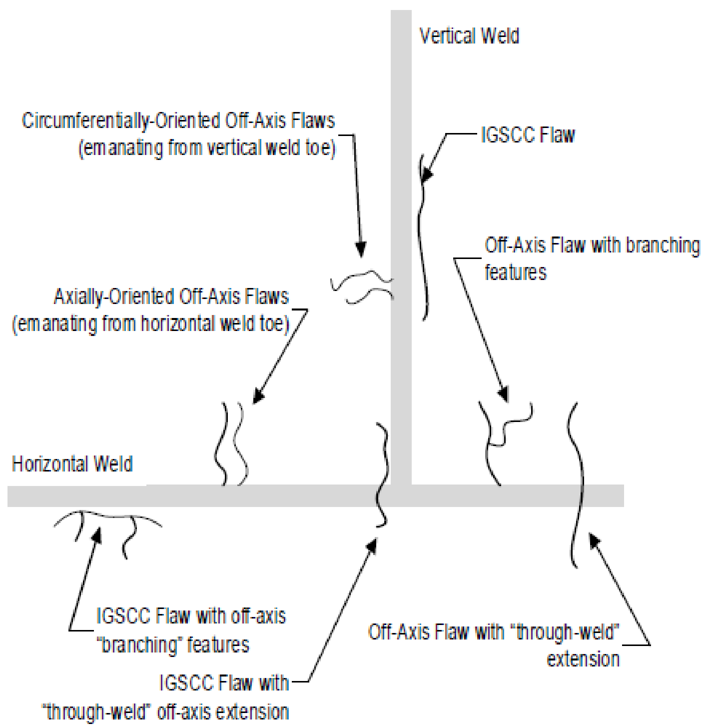

Figure 3-1: Locations and Orientations of BWR Shroud Cracking [14]

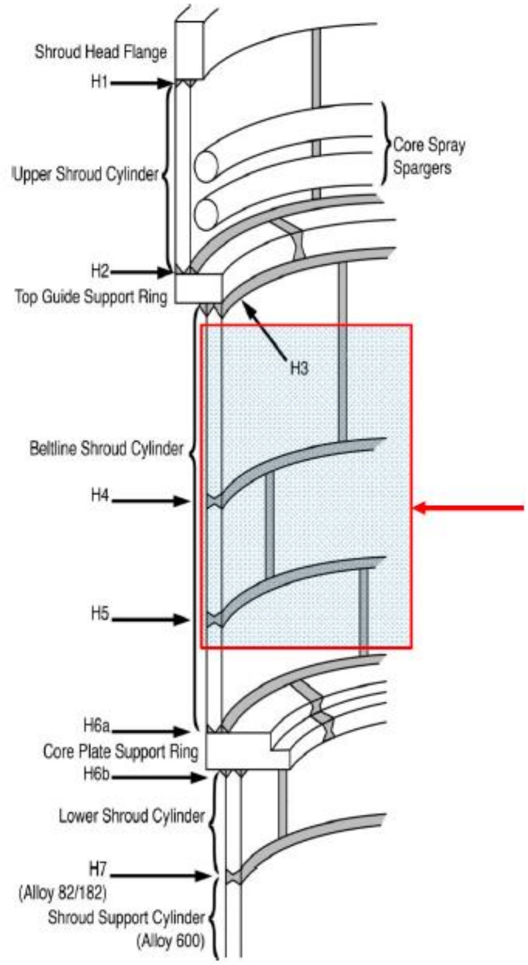

Figure 3-2: Location of BWR Shroud Off-axis Cracking—nearly all off-axis cracking is between weld numbers H3 and H6a, as shown by the arrow. [14]

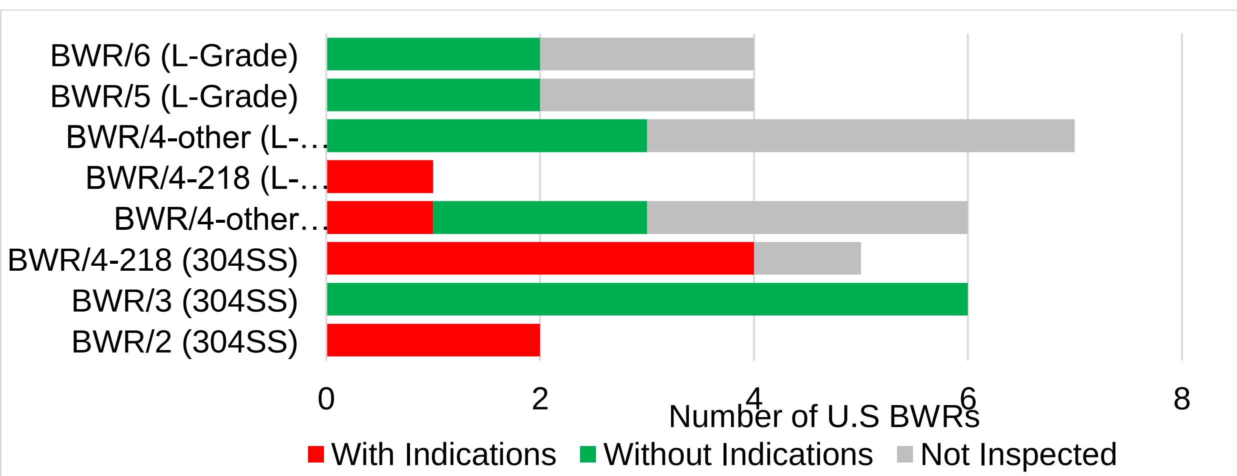

Figure 3-3: Occurrence of off-axis cracking by Plant Design [14]

4 PWR Cracking Experience Summary

Unlike the BWR core shroud experience, observations of cracking in PWR core barrels have been very limited. Two incidents were summarized in the operating experience section of MRP-227-A [7] which occurred in Combustion Engineering-designed PWRs and were due to flow-induced vibration caused failures in the thermal shield. These two events occurred early in plant life, and led to extensive cracks in the core barrels located in the base metal away from the welds. The problem was solved by removing the thermal shield, which eliminated the driving force for the cracks, and drilling holes at the ends of the cracks to blunt the tips and prevent further growth. These cracks have remained in service for over thirty years with no further degradation.

Since the implementation of MRP-227-A [7], multiple enhanced visual (EVT-1) examinations have been conducted on the core barrel welds of both Westinghouse and Combustion Engineering-designed plants [33]. These examinations have included unirradiated welds subject to SCC, such as the upper core barrel flange weld, and irradiated welds in the core beltline that are susceptible to IASCC, such as the Westinghouse lower girth weld. Relevant indications have only been observed in one of those inspections to date. That inspection was of a Combustion Engineering-designed PWR core support barrel and included multiple indications observed by the EVT-1 inspection and confirmed by supplemental volumetric examinations. Plant-specific engineering evaluations in response to this operating experience in accordance with WCAP-17096-NP-A [34] approved methodologies are currently underway. All of the other MRP-227-A inspections of core barrels to date have resulted in no relevant indications.

Historical experience with SCC in austenitic stainless steel exposed to PWR primary water environments was reviewed by Hall and Bamford [35] in 2003 and by Ilevbare et al [2] in 2010. Together, these studies provide extensive documentation of the fact that austenitic stainless steel in PWR primary water environments are not subject to the high incidence of SCC experienced by BWRs. These studies found that 83% of the SCC events occurred in low-flow, occluded, or stagnant locations in the primary system, which are where off-normal chemistry conditions are likely to persist [2]. Only about 17% of reported cracking events occurred in free-flow conditions where normal bulk coolant chemistry is expected conditions, and all of these events were associated with severe cold work. As detailed in Section 2, cold work has been specifically limited in austenitic stainless steel components, which limits the possibility of cold-work induced SCC.

Unlike the experience with BWR core shrouds and other components, environmentally-induced cracking has been very limited in PWR austenitic stainless steel components. This is due to the differences in reactor coolant chemistry between BWR and PWR designs. As described in Section 2, the hydrogen overpressure effectively scavenges all oxygen from the primary coolant system, which creates a low ECP environment in PWR primary water and has prevented SCC degradation of the core barrel to date. Continued operation with a low ECP environment is expected to be an effective strategy for mitigating future SCC in PWR core barrels.

5 Summary and Conclusions

This report has provided a comparison of PWR and BWR Core barrels and shrouds, which are large-diameter austenitic stainless steel structures whose primary purpose is to direct the coolant flow through the reactor vessel core region. The mechanical design criteria and temperatures of operation for these two structures are very similar, but there are major differences in the reactor coolant water chemistry to which they are exposed.

The primary difference is reflected in the electrochemical potential, which measures whether or not the environment is more oxidizing or more reducing. The PWR primary water environment is the most reducing case at approximately -770 mV_{SHE}. This is at least several hundred mV lower than BWR HWC environments and nearly 1000 mV lower than the ECP in a BWR NWC regime. Operation below approximately -230 mV_{SHE} results in mitigation of SCC and IASCC. The beneficial effects of a reducing environment have been explained in Section 2 of this report. The service experience supports these conclusions as well and was reviewed in Sections 3 and 4.

BWR core shrouds, as well as other internals components fabricated from austenitic stainless steels, have experienced significant levels of environmentally-induced cracking, while little cracking has been observed to date in PWR environments and mostly in bolting. Multiple detailed inspections under the requirements of MRP-227 have been completed on PWR core barrels, and only one recent inspection has discovered relevant indications. Since PWRs will operate for their entire licensed lifetimes with low ECP in the primary coolant system due to hydrogen additions, the likelihood of SCC or IASCC initiating in the core barrel is expected to be low.

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MRP Materials Reliability Program ______________________________ MRP 2018-026

Date: September 28, 2018

To: U.S. Nuclear Regulatory Commission
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Mail Stop: 0-12-D2
11555 Rockville Pike
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From: Mike Hoehn II, MRP RIC Chair, Ameren-Missouri
Brian Burgos, Program Manager, EPRI-MRP

Subject: Transmit Initial Industry Responses Regarding EPRI Technical Report MRP-227-Revision 1, "*Materials Reliability Program: Pressurized Water Reactor Internals Inspection and Evaluation Guidelines*" (Project: 0669)

Dear Sir:

This letter is being transmitted to the Nuclear Regulatory Commission (NRC) to support the current NRC safety evaluation of the MRP-227, Revision 1 Reactor Vessel Internals Inspection and Evaluation Guidelines. This letter transmits initial responses to NRC staff's supplemental questions in support of the Nuclear Regulatory Commission Safety Evaluation of MRP-227, Revision 1, as was discussed during a public meeting on 5/23/2018 with NRC staff, and requested by NRC in ML18176A188.

NRC RAI 28

During the May 2018 Materials Information Exchange Meeting, the Electric Power Research Institute, Materials Reliability Program (EPRI-MRP) and the Pressurized Water Reactor Owner's Group (PWROG) Materials Subcommittee made presentations describing recent operating experience with accelerated wear of control rod drive mechanism (CRDM) thermal sleeves (ML18142A395, ML18142A457). This wear has the potential to generate loose parts which could jeopardize control rod insertion.

Describe how this operating experience will be addressed in MRP-227, Rev.1.

Industry Response: The CRDM thermal sleeve degradation is not a new issue with Westinghouse PWRs. The plants most significantly affected have previously planned or conducted inspections based on Technical Bulletin TB-07-2, Rev.3. However, the safety implications were re-evaluated in light of the recent operating experience as discussed in the Materials Information Exchange

Meeting and as communicated in the Part 21 notification. The NSSS OEM has formally notified the industry of the revised evaluation of the degradation by issuing the Westinghouse NSAL (Nuclear Safety Advisory letter) 18-1, dated 7/9/2018, and has contacted the potentially affected plants. This OE was also communicated to PWR plants in April 2018, via EPRI letter MRP 2018-011, dated 4/20/2018. The industry is preparing NEI 03-08 interim guidance associated with the NSSS OEM's recommendations in the NSAL-18-1. The schedule to issue this interim guidance does not match the current schedule of MRP-227-Rev 1 SER. Therefore, if interim guidance is developed, it would be incorporated into MRP-227 Revision 2. It is expected that the NSAL/interim guidance will allow industry to gather more extensive baseline inspection information during the next two outage seasons. Thus, NSAL-18-1 and any future interim guidance will better inform industry for generic incorporation during the MRP-227 Revision 2 update effort.

RAI 29

During the 2018 Materials Information Exchange Meeting, the EPRI MRP reported that, during the spring 2018 outage, a domestic Combustion Engineering (CE) plant identified cracks on the outer diameter (OD) surface of the core barrel in the belt-line elevation using enhanced visual (EVT-1) examination (ML18142A394). EPRI indicated that one crack-like indication was found in base-metal adjacent to the middle-girth weld, which is a primary component in MRP-227-A (ML120170453), and that several crack-like indications were found in base-metal adjacent to the middle-axial weld, which is an expansion component in MRP-227-A. EPRI stated that industry established joint EPRI/PWROG Focus Group similar to Baffle-Former-Bolt Focus Group that was established in 2016, with the intent to provide generic assessment of impact of OE to industry. Describe how this operating experience will be addressed in MRP-227, Rev.1.

Industry Response: Industry has formed a joint EPRI/PWR Owners Group focus group (FG) to evaluate the OE related to a domestic Combustion Engineering plant that identified cracks on the outer diameter(OD) surface of the core barrel in the belt line elevation. The FG is still gathering information related to this OE. The current schedule to complete this review and to issue NEI 03-08 guidance if required does not match the current schedule of MRP-227 Rev.1 SER. For the purposes of maintaining the schedule for the SER and with regard to the OE, the changes shown in the Tables 1 through 4 are proposed for core barrel weld inspections under MRP-227 Revision 1. These changes supplement the prior industry responses associated with RAI # 5.

NRC RAI 30

EPRI's response to the Request for Additional Information (RAI) 8 indicated the interim guidance for baffle-former bolt (BFB) examinations will be incorporated in the final version of MRP-227, Rev. 1. The NRC staff assessment of the BFB interim guidance (ML17310A861) contained the following recommendation regarding the submittal of plant-specific evaluations of BFB subsequent examination interval:

If the table in MRP 2017-009 indicates that the subsequent inspection interval is not to exceed 6 years (e.g., downflow plants with ≥ 3% BFBs with indications or clustering, or upflow plants with ≥ 5% of BFBs with indications or clustering), the plant-specific evaluation to determine a subsequent inspection interval should be submitted to the NRC for information within one year following the outage in which the degradation was found. Any evaluation to lengthen the determined inspection interval or to exceed the maximum inspection interval recommended in MRP-2017-009 should be submitted to the NRC for information at least one year prior to the end of the current applicable interval for BFB subsequent examination. This recommendation should be incorporated into the final NRC-approved version of MRP-227, Rev. 1.

Please confirm that this NRC staff recommendation will be included in MRP-227, Rev. 1, or another NRC-approved industry guidance document, such as WCAP-17096-NP-A (ML16279A320) or provide a basis for not doing so.

Industry response: One part of this recommendation is included in the industry's interim guidance for item W-ID-7 associated with WCAP-17096-NP-A, which is documented in the PWR Owners Group report PWROG-17071-NP Section 2.3.2. This PWROG-17071-NP report was provided to the NRC as requested in the 5/23/2018 public meeting. (Refer to NRC ADAMS accession number ML18204A165 and ML18204A166.) Plans are to revise WCAP-17096-NP-A in 2019. The second part of this recommendation will be addressed in the revision to WCAP-17096-NP-A.

If you have any questions, please contact either Tim Wells-Southern Nuclear Co. (tgwells@southernco.com or 205-992-7460), Kyle Amberge-EPRI-MRP (kamberge@epri.com or 980-266-2646), or myself (mhoehn@ameren.com, (314) 225-1543).

Sincerely,

Mike Hoehn II	Brian Burgos
MRP RIC Chair	MRP Program Manager
Ameren-Missouri	EPRI

Table 1: Proposed Entry Changes for Table 4-2 (RAI #29)
CE Plants Primary Components

Primary Item	Applicability	Effect (Mechanism)	Expansion Link (Note 1)	Examination Method/Frequency (Note 1)	Examination Coverage
C5.Core Support Barrel Assembly Upper flange weld (UFW)	All plants	Cracking (SCC)	Upper Girth Weld (UGW), Lower Girth/Flange Weld (LGW/LFW), Upper Axial Welds (UAW), Lower core support beams	Enhanced visual (EVT-1) examination no later than 2 refueling outages from the beginning of the license renewal period. Subsequent examinations on a ten-year interval.	100% of the accessible weld length of one side of the UFW and 3/4" of adjacent base metal shall be examined. (Note 5) See Figure 4-29.
C6.Core Support Barrel Assembly Middle Girth Weld (MGW)	All plants	Cracking (SCC, IASCC) Aging Management (IE)	Middle Axial Weld (MAW), Lower Axial Weld (LAW)	Enhanced visual (EVT-1) examination no later than 2 refueling outages from the beginning of the license renewal period. Subsequent examinations on a ten-year interval.	100% of the accessible weld length of the OD of the MGW and 3/4" of adjacent base metal shall be examined. (Note 5) See Figure 4-29.

Notes to Table 4-2:

1. Examination acceptance criteria and expansion criteria for the CE components are in Table 5-2.
2. A minimum of 75% of the total core shroud bolts population (examined + unexamined), including coverage consistent with the Expansion criteria in Table 5-2, must be examined for inspection credit.
3. A minimum of 25% of the total population of core support column welds
4. If evidence of distortion is detected by visual exam, consideration should be given to making supplementary measurements (minimum of 3 to 5, depending on extent of observed condition) of gap opening between the upper and lower core shroud segments.
5. Examination coverage requires a minimum of 75% of the length of either the ID or the OD of the weld being examined.

Table 2: Proposed Entry Changes for Table 4-1 (RAI #29)
CE Plants Expansion Components

Expansion Item	Applicability	Effect (Mechanism)	Primary Link (Note 1)	Examination Method/Frequency (Note 1)	Examination Coverage
Core Support Barrel Assembly C5.1.Lower Girth Weld (LGW)	All plants	Cracking (SCC, Fatigue)	C5.Upper Flange Weld (UFW)	Enhanced visual (EVT-1) examination. Re-inspection every 10 years following initial inspection.	100% of the accessible weld length of the OD of the LGW and 3/4" of adjacent base metal shall be examined. (Note 3) See Figure 4-29.
Core Support Barrel Assembly C5.2.Upper Girth Weld (UGW) and C5.3.Upper Axial Weld (UAW)	All plants	Cracking (SCC) Aging Management (IE)	C5.Upper Flange Weld (UFW)	Enhanced visual (EVT-1) examination. Re-inspection every 10 years following initial inspection.	100% of the accessible weld length of one side of the UGW and UAW and 3/4" adjacent base metal shall be examined. (Notes 2 and 3) See Figure 4-29.
Core Support Barrel Assembly C6.1.Middle Axial Weld (MAW), C6.2.Lower Axial Weld (LAW)	All plants	Cracking (SCC, IASCC) Aging Management (IE)	C6.Middle Girth Weld (MGW)	Enhanced visual (EVT-1) examination Re-inspection every 10 years following initial inspection.	100% of the accessible weld length of the OD of the MAW and LAW and 3/4” adjacent base metal shall be examined. (Note 3) See Figure 4-29.

Table 2: Proposed Entry Changes for Notes to Table 4-5:
1. Examination acceptance criteria and expansion criteria for the CE components are in Table 5-2.
2. Examination coverage requires examination of either the ID or the OD of the weld.
3. Examination coverage requires a minimum of 75% of the weld length for either the ID or the OD of the weld being examined.

Table 3: Proposed Entry Changes for Table 4-2 (RAI #29)
Westinghouse Plants Primary Components

Primary Item	Applicability	Effect (Mechanism)	Expansion Link (Note 1)	Examination Method/Frequency (Note 1)	Examination Coverage
W3.Core Barrel Assembly Upper flange Weld (UFW)	All plants	Cracking (SCC)	Upper girth weld (UGW), lower flange weld (LFW), upper axial welds (UAW), and lower support forging or casting	Enhanced visual (EVT-1) examination, no later than 2 refueling outages from the beginning of the license renewal period and subsequent examination on a ten-year interval.	100% of the accessible weld length of one side of the UFW and 3/4" of adjacent base metal shall be examined. (Note 8) See Figure 4-13.
W4.Core Barrel Assembly Lower girth weld (LGW)	All plants	Cracking (SCC, IASCC, Fatigue), Aging Management (IE)	Upper core plate, Lower support column bodies (cast, non-cast), middle axial welds (MAW), lower axial welds (LAW)	Enhanced visual (EVT-1) examination, no later than 2 refueling outages from the beginning of the license renewal period and subsequent examination on a ten-year interval.	100% of the accessible weld length of the OD of the LGW and 3/4" of adjacent base metal shall be examined. (Note 8) See Figure 4-13.

Notes to Table 4-3:

1. Examination acceptance criteria and expansion criteria for the Westinghouse components are in Table 5-3.
2. A minimum of 75% of the total identified sample population must be examined.
3. A minimum of 75% of the total bolt population (examined + unexamined), including coverage consistent with the Expansion criteria in Table 5-3, must be examined for inspection credit.
4. A minimum of 50% of the total weld length (examined + unexamined), including coverage consistent with the Expansion criteria in Table 5-3, must be examined from either the inner or outer diameter for inspection credit.

5. The lower core barrel flange weld may be alternatively designated as the core barrel-to-support plate weld in some Westinghouse plant designs.
6. Void swelling effects on this component are managed through management of void swelling on the entire baffle-former assembly.
7. In WCAP-17451-P Revision 1 the baseline examination schedule has been adjusted for various CRGT designs, the extent of individual CRGT examination modified, and flexible subsequent examination regimens correlating to initial baseline sample size, accuracy of wear estimation and examination results. Initial inspection prior to the license renewal period may be required.
8. Examination coverage requires a minimum of 50% of the length of either the ID or the OD of the weld being examined.
9. Baffle-former bolt inspection includes inspection of the corner plate bolts when applicable.

Table 4: Proposed Entry Changes for Table 4-3 (RAI #29)
Westinghouse Plants Expansion Components

Expansion Item	Applicability	Effect (Mechanism)	Primary Link (Note 1)	Examination Method/Frequency (Note 1)	Examination Coverage
Core Barrel Assembly W3.1.Upper Girth Weld (UGW)	All plants	Cracking (SCC)	W3.Upper Core Barrel Flange Weld (UFW)	Enhanced visual (EVT-1) examination. Re-inspection every 10 years following initial inspection.	100% of the accessible weld length of one side of the UGW and 3/4" of adjacent base metal shall be examined (Notes 2 and 3). See Figure 4-13.
Core Barrel Assembly W3.2.Upper Axial Weld (UAW)	All plants	Cracking (SCC)	W3.Upper Core Barrel Flange Weld (UFW)	Enhanced visual (EVT-1) examination. Re-inspection every 10 years following initial inspection.	100% of the accessible weld length of one side of the UAW and 3/4" of adjacent base metal shall be examined (Notes 2 and 3). See Figure 4-13.
Core Barrel Assembly W3.3.Lower Flange Weld (LFW)	All plants	Cracking (SCC)	W3.Upper Core Barrel Flange Weld (UFW)	Enhanced visual (EVT-1) examination. Re-inspection every 10 years following initial inspection.	100% of the accessible weld length of the OD surface of the LFW and 3/4" of adjacent base metal shall be examined (Note 3). See Figure 4-13.

Table 4: Proposed Entry Changes for Table 4-6 (RAI #29)
Westinghouse Plants Expansion Components

Expansion Item	Applicability	Effect (Mechanism)	Primary Link (Note 1)	Examination Method/Frequency (Note 1)	Examination Coverage
Core Barrel Assembly W4.2.Middle Axial Welds (MAW) and W4.3.Lower Axial Welds (LAW)	All plants	Cracking (SCC, IASCC) Aging Management (IE)	W4.Lower Girth Weld (LGW)	Enhanced visual (EVT-1) examination. Re-inspection every 10 years following initial inspection	100% of the accessible weld length of the OD of the MAW and LAW and 3/4" of adjacent base metal shall be examined (Notes 3 and 4). See Figure 4-13.

Notes to Table 4-6:

1. Examination acceptance criteria and expansion criteria for the Westinghouse components are in Table 5-3.
2. Examination coverage requires examination of either the ID or the OD of the weld.
3. A minimum coverage of 75% of the weld length on the surface being examined shall be achieved; however, for welds with limited access (Note 4), a minimum examination coverage of 50% of the weld length on the surface being examined shall be achieved.
4. Accessibility to the MAW and LAW may be limited by the thermal shield or neutron panels—no disassembly to achieve higher weld length coverage is required.

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