WCAP-17096-NP-A
Revision 3

August 2023

# Reactor Internals Acceptance Criteria Methodology and Data Requirements

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**WCAP-17096-NP-A**
**Revision 3**

# Reactor Internals Acceptance Criteria Methodology and Data Requirements

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**August 2023**

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Work Performed under PA-MSC-1567

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NRC Transmittal Letter, “NRC Transmittal of the Final Safety Evaluation for Pressurized Water Reactor Owners Group (PWROG) Topical Report (TR) WCAP-17096-NP, Revision 3, ‘Reactor Internals Acceptance Methodology and Data Requirements’,” July 24, 2023…… page iv

NRC Safety Evaluation, “Final Safety Evaluation by the Office of Nuclear Reactor Regulation for the Pressurized Water Reactor Owners Group Topical Report WCAP-17096-NP, Revision 3, ‘Reactor Internals Acceptance Criteria Methodology and Data Requirements,’ EPID L-2019-TOP-0049,” July 24, 2023…… page vi



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| | |
|---|---|
| **From:** | Luke Haeg |
| **To:** | Olinski, Dewey C. |
| **Cc:** | Holderbaum, Chad M.; Andrachek, James; Gerond George; Leslie Fields |
| **Subject:** | FW: NRC Transmittal of Final Safety Evaluation for WCAP-17096-NP, Revision 3 |
| **Date:** | Monday, July 24, 2023 11:52:00 AM |

Mr. Dewey Olinski, PE
Executive Director
PWR Owners Group, Program Management Office
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SUBJECT: NRC Transmittal of the Final Safety Evaluation for Pressurized Water Reactor Owners Group (PWROG)
Topical Report (TR) WCAP-17096-NP, Revision 3, "Reactor Internals Acceptance Criteria Methodology and Data Requirements"

Dear Mr. Olinski,

By letter dated July 31, 2019 (Agencywide Documents Access and Management System Accession No. (ADAMS No.) ML19218A185), as supplemented by letters dated April 8, 2021 (ADAMS No. ML22187A141), and January 14, 2022 (ADAMS No. ML22187A149), the Pressurized Water Reactor Owners Group (PWROG) submitted Topical Report (TR) WCAP-17096-NP, Revision 3, "Reactor Internals Acceptance Criteria Methodology and Data Requirements" (ADAMS No. ML19218A179), to the U.S. Nuclear Regulatory Commission (NRC) for review and approval. By email dated September 20, 2022 (ADAMS No. ML22266A280), the NRC staff issued its draft safety evaluation (SE) for WCAP-17096-NP, Revision 3.

The PWROG provided comments on the draft SE via letter (ADAMS No. ML23009A589) on January 6, 2023. A copy of the final SE for TR WCAP-17096-NP, Revision 3, has been uploaded to box.com and has been placed in ADAMS under ML23123A088.

The NRC staff concludes that the PWROG TR is acceptable for referencing in licensing applications to the extent specified and discussed in the SE. The final SE defines the basis of our acceptance of the TR. Our acceptance applies only to material provided in the subject TR. In accordance with the guidance provided on the NRC website, we request that PWROG publish an accepted version of the TR within three months of receipt of the date of this email. The accepted version shall incorporate this email and the final SE after the title page.

Along with the final SE, this email will be placed in ADAMS under package ML23205A074 and will be declared as an Official Agency Record for public availability.

Please be informed that the accepted versions of this TR must contain historical review information, including NRC requests for additional information (RAI) questions and the associated responses. The accepted versions shall also include a "-A" (designating approved) following the TR identification symbol.



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

1. The RAI questions and responses can be included as an Appendix to the accepted version.
2. The RAI questions and 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 RAI questions 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, PWROG 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, please contact Leslie Fields at 301-415-1186 or via email at leslie.fields@nrc.gov.

Luke Haeg
Senior Project Manager
NRR/DORL/LLPB
U.S. Nuclear Regulatory Commission
301-415-0272



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# FINAL SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

# FOR THE PRESSURIZED WATER REACTOR OWNERS GROUP

# TOPICAL REPORT WCAP-17096-NP, REVISION 3,

# "REACTOR INTERNALS ACCEPTANCE CRITERIA METHODOLOGY

# AND DATA REQUIREMENTS"

# EPID L-2019-TOP-0049

## 1.0 INTRODUCTION

By letter dated July 31, 2019 (Reference 1), as supplemented by letters dated April 8, 2021 (Reference 2), and January 14, 2022 (Reference 3), the Pressurized Water Reactor Owners Group (PWROG) submitted Topical Report (TR) WCAP-17096-NP, Revision 3, "Reactor Internals Acceptance Criteria Methodology and Data Requirements" (Reference 4), to the U.S. Nuclear Regulatory Commission (NRC) for review and approval.

The purpose of the subject TR is to provide methodologies for demonstrating PWR reactor vessel internals (RVI) integrity throughout the life of the plant, which includes the extended period authorized by license renewal in accordance with Title 10 of the *Code of Federal Regulations* (10 CFR) Part 54, "Requirements for Renewal of Operating Licenses for Nuclear Power Plants." The updated acceptance criteria methodology and data requirements for PWR RVI components in the subject TR are based on the PWROG's effort to update the PWR RVI guidelines in WCAP-17096-NP-A, Revision 2, "Reactor Internals Acceptance Criteria Methodology and Data Requirements," based on changes made to the augmented inspection and evaluation (I&E) criteria for PWR RVI components in the Electric Power Research Institute (EPRI) Non-Proprietary Report No. 3002017168, "Materials Reliability Program [MRP] 227: Pressurized Water Reactor Internals Inspection and Evaluation Guidelines," Revision 1 – NRC-approved or (-A) (Reference 5).

In general, the NRC staff's review of the subject TR evaluated how it reinforces NRC regulations and guidelines as noted within this safety evaluation (SE). Table 1, "Historical Timeline of NRC-Approved Guidance for Review of RVI Components," below, provides a historical account of NRC-approved documents, approval dates, and corresponding agency records available in the NRC Agencywide Documents Access and Management System (ADAMS). The NRC staff referred to these documents to verify updates to aging management criteria and to confirm the resolution of requests for additional information (RAIs) in previously processed TRs that were reviewed by the NRC.



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**Table 1 - Historical Timeline of NRC-Approved Guidance for Review of RVI Components**

| Document Title | Approval Date | Agency Record | Reference No. |
|---|---|---|---|
| NUREG-1801, Revision 2 (GALL Report) | December 2010 | ML103490041 | Reference 6 |
| SE (Revision 0) for TR MRP-227, Revision 0 | June 22, 2011 | ML111600498 | Reference 7 |
| SE (Revision 1) for TR MRP-227, Revision 1 | December 16, 2011 | ML11308A770 | Reference 8 |
| TR MRP-227-A, Revision 0 | December 2011 | ML12017A194 | Reference 9 |
| NRC's Endorsement Letter MRP-227-A, Revision 0 | February 3, 2012 | ML120270374 | Reference 10 |
| TR MRP-227-A, Interim Guidance | February 18, 2014 | ML14274A372 | Reference 11 |
| SE for TR WCAP-17096-NP, Revision 2 | May 3, 2016 | ML16061A243 | Reference 12 |
| TR WCAP-17096-NP-A, Revision 2 | August 31, 2016 | ML16279A320 | Reference 13 |
| NRC Approval Letter WCAP-17096-NP-A, Revision 2 | January 3, 2017 | ML16271A001 | Reference 14 |
| NUREG-2191, Vol. 1 (GALL-SLR Report (Vol. 1) | February 7, 2017 | ML16274A389 | Reference 15 |
| NUREG-2191, Vol. 2 (GALL-SLR Report (Vol. 2) | February 7, 2017 | ML16274A399 | Reference 16 |
| SE for TR MRP-227, Revision 1 | April 25, 2019 | ML19081A001 | Reference 17 |
| TR MRP-227, Revision 1-A | June 22, 2020 | ML20175A112 | Reference 18 |
| NRC's SLR-Interim Staff Guidance SLR-ISG-2021-01-PWRVI | January 8, 2021 | ML20217L203 | Reference 19 |
| Federal Register Notice (FRN) for SLR-Interim Staff Guidance SLR-ISG-2021-01-PWRVI | January 13, 2021 | ML20217L212 | Reference 20 |

This SE provides a summary of NRC staff's review and evaluation of the methodology and acceptance criteria for the WCAP-17096-NP, Revision 3, "Reactor Internals Acceptance Criteria Methodology and Data Requirements." The NRC staff performed the review based on a comparison of WCAP-17096-NP, Revision 3, to previously approved guidance listed in Table 1, above.

The scope of the NRC staff's review in this SE included an assessment of a given RVI component provided in Appendices A, C, or E of the subject TR where the component identification (ID) item assessment was the subject of an RAI or was identified as needing further staff clarifications. The NRC staff's evaluation does not analyze any new component ID item assessments for specified RVI components in Appendix A, C, or E of the subject TR (where the given component has been defined as a new "Primary" or "Expansion" category component as described in MRP-227, Revision 1-A, and where the PWROG has developed a new component ID item assessment for the equivalent component provided in Appendix A, C, or E of the subject TR).

For the review of the subject TR, the NRC staff prepared the SE input based on the following assessment factors:

- The PWROG's bases for closure of NRC's conditions that were previously issued in NRC's SE for WCAP-17096-NP, Revision 2, dated May 3, 2016 (Reference 12), and the PWROG's proposed resolution as defined in Section 2.2 of the subject TR,

- The PWROG's basis in the subject TR for addressing Applicant/Licensee Action Item-01 (A/LAI-01) that was previously issued in NRC's SE dated April 25, 2019 (Reference 17) for MRP-227, Revision 1-A,



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- Changes that needed to be incorporated into the subject TR as a result of changes made to MRP-227, Revision 1-A, including supplemental EPRI MRP reports, PWR methodologies, or guidelines that apply to the RVI components, and changes that were subsequently issued to the PWROG's issuance of WCAP-17096-NP, Revision 2-A,

- Items discussed in the subject TR that were not changed from those in WCAP-17096-NP, Revision 2, for which the NRC staff determined that further clarifications were needed related to the updated guidelines in MRP-227, Revision 1-A,

- Items discussed in the subject TR where the NRC staff determined it needed RAIs, including, but not limited to, RAIs on the disposition of recent PWR RVI operating experience (OE) and RAIs on component methodologies/data requirements that were referenced in analysis involving probabilistic criteria, fracture toughness, or crack growth criteria not yet approved by NRC, and

- Methodologies presented in the subject TR that continue to be referenced to as "*implied but yet-to-be developed,*" industry analysis methodologies, or inspection standards.

Additionally, for the factors presented in this section, the NRC staff made a determination to issue a limitation or condition based on the methodology and criteria within (or outside) the scope of the subject TR such as:

- NRC staff's determination that the PWROG's methodology in the subject TR places the responsibility for addressing a component acceptance criterion/data analysis criterion to a licensee that may be implementing the report for its licensing or design basis,

- NRC staff's determination that the PWROG places a given assessment criterion or PWROG-identified additional actions for a given component in the subject TR (including those for managing specific aging effects or mechanisms) outside the scope of the subject TR,

- NRC staff's determination that the PWROG-specified acceptance criteria methodology or data requirement basis for a given component ID item assessment in Appendix A, C, or E of the subject TR was based on the PWROG's reference to an "*implied but yet-to-be developed*" industry analysis methodology or inspection standard,

- NRC staff's determination that the PWROG's component methodology or data requirement may be relying on industry methodologies not approved by NRC, including those utilizing probabilistic or risk-informed analysis methods, and



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- NRC's staff's determination that a limitation or condition may be necessary for the subject TR based on the consequences of a component or generic OE not accounted for or bounded by the component ID item assessment bases used to develop MRP-227, Revision 1-A.

Finally, for the review of the subject TR, the NRC staff did not focus its review on unchanged items or administrative changes (edits) made in the subject TR that the NRC staff considered to be solely administrative in nature.

## 2.0 REGULATORY EVALUATION

### 2.1 NRC Regulations and Licensing Requirements

Title 10 CFR, Part 54, "Requirements for Renewal of Operating Licenses for Nuclear Power Plants," sets forth the requirements for renewal of commercial nuclear power plant operating licenses. A license renewal application (LRA) or a subsequent license renewal application (SLRA) submitted pursuant to 10 CFR Part 54 must contain the information specified in 10 CFR 54.19, "Contents of application—general information."

The regulation in 10 CFR 54.21(3) requires that, for each structure and component determined to be subject to an aging management review (AMR), the license renewal applicant demonstrates that the effects of aging will be adequately managed so that the intended function(s) will be maintained consistent with the current licensing basis (CLB[1]) for the relevant period of extended operation.

In the subject TR, the regulatory basis for the PWROG's acceptance criteria methodology and data requirements are established from 10 CFR Part 54 and the terms and conditions in renewed licenses or subsequent renewed licenses for the initial or subsequent periods of extended operation for PWR-designed power plants. Specifically, the first[2] renewed or subsequent renewed operating licenses for PWR-designed plants include specific terms and conditions that require the NRC licensees of nuclear power plant facilities to implement their aging management programs (AMPs) and time-limited aging analyses (TLAAs) for systems, structures, and components (SSCs) that were within the scope of license renewal and subject to either an AMR or a TLAA in the NRC's review of the LRA or SLRA for the facility. The first renewed or subsequent renewed operating licenses for a given PWR-design light water reactor facility may also include terms and conditions that permit the NRC licensee to make changes to their AMPs (including the PWRVI AMP or equivalent AMP for the RVI components) based on implementation of the 10 CFR 50.59 process.

The regulation of 10 CFR 54.35 establishes (in part) the Commission's requirement that the holders of a renewed or subsequent renewed operating license will continue to comply with the requirements of 10 CFR Part 54 during the licensed term of a renewed or subsequent renewed operating license.

[1] As defined in 10 CFR 54.3, *current licensing basis* (CLB) is the set of NRC requirements applicable to a specific plant and a licensee's written commitments for ensuring compliance with and operation within applicable NRC requirements and the plant-specific design basis (including all modifications and additions to such commitments over the life of the license) that are docketed and in effect.

[2] For the purposes of this SE, the initial renewed operating license is known as the first renewed license issued under 10 CFR Part 54.



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## 2.2 NRC Guidance and Industry Guidelines

For PWR RVI components that are required to be within the scope of license renewal in accordance with 10 CFR 54.4(a) and subject to an AMR in accordance with 10 CFR 54.21(a)(1), NRC staff developed guidance for PWR RVI AMPs in the Generic Aging Lessons Learned for Subsequent License Renewal (GALL-SLR) Report – Final Report (NUREG-2191, Volumes 1 and 2) (References 15 and 16), AMP XI.M16A, “PWR Vessel Internals”, in accordance with NRC’s Interim Staff Guidance (ISG), “Updated Aging Management Criteria for Reactor Vessel Internal Components for Pressurized-Water Reactors” (SLR-ISG-2021-01-PWRVI or SLR-ISG) (References 19 and 20). To assist NRC-licensed PWR facilities in addressing the program element criteria established in GALL-SLR Report AMP XI.M16A, the EPRI MRP developed a generic, sampling based I&E methodology for managing the effects of aging in PWR RVI components, with the current NRC-approved version of the guidance established in MRP-227, Revision 1-A.

The PWROG’s evaluation methodology, acceptance criteria, and data analysis criteria in the subject TR are linked to the component I&E criteria established in MRP-227, Revision 1-A. In the SLR-ISG, the NRC updated Section AMP XI.M16A to allow the AMP (through the subsequent period of extended operation) to be based on the EPRI’s version of MRP-227, Revision 1-A, as subject to an 80-year gap analysis to account for potential adjustments that might need to be made to I&E criteria in MRP-227, Revision 1-A during a subsequent renewed licensing period. The SLR-ISG allows relevant supplemental methodologies, guidelines, or reports to be included within the scope of a licensee’s PWR vessel internals AMP, particularly if the reports, guidelines, or methodologies were determined to be applicable to the CLB for the facility. This includes an NRC licensee’s potential use of the subject TR as a supplemental methodology, even though it is not specifically cited or referenced as a supplemental methodology in the version of GALL-SLR AMP XI.M16A in NUREG-2191, Volume 2, or in the updated version of this AMP in the SLR-ISG.

The industry guidelines referenced in this SE, Nuclear Energy Institute (NEI) NEI 03-08 (Reference 22), which include the recommended classification of each material issue as “Mandatory,” “Needed,” or “Good Practice,” is dependent on the relative level of importance of the issue and associated recommended action. The NRC staff notes that the mention of these types of criteria in the SE are not being established, referenced, or cited as NRC requirements, but rather are established by the NEI for NRC licensee-implemented programs (including PWR VIPs or the equivalent named in MRP-227 program) in alignment with the NEI “materials management” industry initiatives of NEI 03-08.

# 3.0 TECHNICAL EVALUATION

This technical evaluation section documents the NRC staff's evaluation of the subject TR against the relevant criteria identified in Section 2.0.

## 3.1 Overview of Topical Report WCAP-17096, Revision 3

The main objective of the subject TR is to identify consistent, industry-wide analytical methodologies and data requirements for developing: (1) acceptance criteria for the “Primary” and “Expansion” components identified in the MRP-227, Revision 1-A, and (2) evaluation procedures for facilities to assess potential safety and functional impacts degradation in components with observed relevant conditions.



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3.2 Applicability of Topical Report WCAP-17096, Revision 3, to the Current Licensing Basis

The "Limitations and Conditions" defined in Table A-1 of Attachment A to this SE provide details on the use and generic applicability of the subject TR. The subject TR is dependent on the following factors: (1) the type of plant operating license (i.e., original, first renewed, or subsequent renewed license) held by the NRC-licensed facility which owns the specified PWR unit, (2) the type of AMP (i.e., plant-specific AMP versus generic MRP-227-based AMP) being applied to the PWR RVI components, (3) the type of generic EPRI MRP-based program being applied as the basis for the AMP in the CLB, and (4) the version of MRP-227 being applied to the program.

"Limitations and Conditions" are additional restrictions imposed by the NRC staff to further frame the scope of applicability of a TR and identify any additional plant-specific actions that will be needed to support the NRC staff's review of a request to implement the subject TR. "Limitations and Conditions" identify the items that will need to be addressed in subsequent individual plant applications that will reference the subject TR. The NRC staff has enclosed a summary table in Attachment A to this SE named, "Table A-1, Summary of Limitations and Conditions" for Topical Report WCAP-17096, Revision 3. The table provides a brief synopsis and numerical order of limitation and condition items (e.g., formatted as L-01, C-01, etc.) for the subject TR.

As stated in Section 1.0 of this SE, the PWROG's updated acceptance criteria methodology and data requirements for PWR RVI components in the subject TR presents an update to the previous guidelines in WCAP-17096-NP, Revision 2. The updates are based on changes made to augmented I&E criteria for PWR RVI components in MRP-227, Revision 1-A from the prior version of the EPRI MRP methodology in MRP-227-A. Therefore, the criteria in the subject TR applies to NRC licensee RVI AMPs and similarly applies to the EPRI MRP I&E methodology in MRP-227, Revision 1-A.

Also, the NRC staff noted that, in Section 1 of the subject TR, the PWROG included a statement[3] to indicate that the methodology and component criteria in the subject TR may be used in conjunction with the prior EPRI MRP I&E methodology in MRP-227-A.

The NRC staff determined that this could potentially create an inconsistency with the EPRI MRP timing criterion for converting an NRC licensee's PWR RVI AMP over to the methodology in MRP-227, Revision 1-A. Specifically, NRC staff noted that, in Section 7 of MRP-227, Revision 1-A, the EPRI MRP establishes an NEI 03-08 "Needed" criterion that recommends a programmatic AMP conversion to MRP-227, Revision 1-A, by the appropriate time identified in the MRP. Therefore, for NRC licensees that implement EPRI MRP-227 site-specific AMPs as part of their NEI 03-08 initiatives, the licensee's AMP would follow the EPRI MRP's implementation criteria in Section 7 of MRP-227 Revision 1-A that called for conversions of program over to the version of MRP-227 in Revision 1-A of the report (by January 1, 2021). The NRC staff addressed this apparent inconsistency in RAI-01 in the NRC's RAI transmittal, dated September 21, 2020 (Reference 23).

In the PWROG's response to RAI-01 (References 24 and 25), it stated that the subject TR is intended to be a companion document to MRP-227 and the subject TR is in alignment with the current version of NEI 03-08 (References 21 and 22). Consequently, the PWROG amended Reference 36 cited in the subject TR to include the updated version of MRP-227 Revision 1-A. Given that the subject TR was updated to account for changes made in MRP-227, Revision 1-A, the NRC staff finds that the PWROG's explanation and its revision to Reference 36 in the subject TR to be acceptable. The NRC

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



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licensees following these MRP-227, Revision 1-A guidelines would convert their MRP-227-based RVI management program over to version MRP-227, Revision 1-A, by the appropriate time identified in the guidance. Therefore, RAI-01 is resolved.

The NRC staff also verified that in Section 1.1 of the subject TR, the PWROG added new information to provide the specific basis for implementing the methodology in the subject TR in alignment with: (1) the three implementation categories known as “Mandatory,” “Needed,” or “Good Practice” defined in NEI 03-08, (2) the implementation protocols established in Section 7 of MRP-227, Revision 1-A, and (3) “Needed” criterion for applicable components as specified in Section 7.0 of MRP-227, Revision 1-A. Specifically, the PWROG states that the methodology in the subject TR does not include any specific NEI 03-08 “Mandatory,” “Needed,” or “Good Practice” criteria, other than those needed to clarify that the methodologies defined in the subject TR which can be used to meet the NEI 03-08 “Needed” criterion specified in Section 7.5 of MRP-227, Revision 1-A.

In general, the NRC staff concluded that the PWROG’s rationale in this regard is acceptable because Section 7 of MRP-227, Revision 1-A permits an NRC licensee to apply the methodology in the subject TR as one of the options that the licensee may have for evaluating the RVI component inspection results. Specifically, Section 7.5 of MRP-227, Revision 1-A establishes the following NEI 03-08 “Needed” criterion for these types of AMPs:

> ***“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 17096-NP-A or equivalent method)."*

The NRC’s updated version of GALL-SLR AMP XI.M16A in the SLR-ISG also accounts for this possibility. Therefore, the NRC staff does not specifically reference the subject TR methodology or the previous methodology in WCAP-17096-NP, Revision 2, in any of the program elements of GALL-SLR AMP XI.M16A (including the updated version of the AMP in the SLR-ISG). As such, the evaluation methodology, acceptance criteria, and data analysis requirements in the subject TR (or in WCAP-17096-NP, Revision 2) only provide one way of demonstrating conformance with the “monitoring and trending” element criteria in the GALL/GALL-SLR AMP XI.M16A or with the NEI “Needed” criterion referenced in Section 7.5 of the MRP-227, Revision 1-A.

### 3.2.1 Licensing Considerations – Comanche Peak, Diablo Canyon, and Watts Bar

L-01(a) (WCAP-17096-NP, Revision 3):
For PWR plants with an original operating license in a pre-submittal phase for first-time license renewal or currently undergoing NRC review of a first-time license renewal application, NRC licensees may use the subject TR if they are implementing AMPs in accordance with MRP-227, Revision 1-A.

During the time of NRC’s review of the subject TR, all NRC licensees with PWR units had received either a first renewed or a subsequent renewed operating license for their PWR units, except for Comanche Peak Steam Electric Station, Units 1 and 2 (Comanche Peak), Diablo Canyon Power Plant, Units 1 and 2 (Diablo Canyon), and Watts Bar Nuclear Plant, Units 1 and 2 (Watts Bar). Therefore, because the Comanche Peak, Diablo Canyon, and Watts Bar original operating licenses did not include aging management terms and conditions discussed in Section 2.1 of this SE, the



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limitation above has been identified and needs to be addressed in subsequent individual plant licensing applications when referencing the subject TR.

The NRC licensees for Comanche Peak, Diablo Canyon, and Watts Bar may opt to implement (on their own initiative) an RVI AMP for their units based on the EPRI I&E criteria in MRP-227, Revision 1-A. The methodology in the subject TR may be applied as part of the AMP regulatory basis for these units, subject to the limitations and conditions of this SE as defined in Table A-1 of Attachment A to this SE which applies to the Westinghouse (W)-designed PWRs.

3.2.2 Holders of First Renewed Operating Licenses

L-01(b) (WCAP-17096-NP, Revision 3):
For PWR plants with first renewed operating licenses, NRC licensees may use the subject TR if they are implementing AMPs in accordance with MRP-227, Revision 1-A.

NRC licensees holding their first renewed operating license for their PWR units may apply the acceptance criteria methodology and data requirements as defined in the subject TR (subject to the limitations and conditions when referencing the subject TR as documented in this SE) where the NRC's safety evaluation report (SER) for the first license renewal references an NRC-approved RVI AMP that is either: (1) based on MRP-227, Revision 1-A, or (2) based on MRP-227-A that has been converted over to MRP-227, Revision 1-A, after the date of issuance of the NRC SER for the license renewal application. This may be used to satisfy the NEI 03-08 "Needed" criterion as referenced in Section 7.5 of the MRP-227, Revision 1-A. Therefore, the limitation above has been identified and needs to be addressed in subsequent individual plant licensing applications when referencing the subject TR.

3.2.3 Holders of Subsequent Renewed Operating Licenses

L-01(c) (WCAP-17096-NP, Revision 3):
For PWR plants with subsequent-renewed operating licenses, NRC licensees may use the subject TR only for those RVI components whose I&E criteria for the components are still consistent with MRP-227, Revision 1-A, and have not been modified by the results of an 80-year RVI gap analysis or a version of MRP-227 covering an 80-year aging assessment period.

For NRC licensees holding subsequent renewed operating licenses of their PWR units, where the NRC SER for the SLR application references an NRC-approved RVI AMP in accordance with MRP-227, Revision 1-A (as supplemented by the 80-year gap analysis as defined in SLR-ISG), or alternatively, an EPRI MRP-based AMP that is based on an NRC-approved 80-year version of the MRP-227 report (e.g. MRP-227, Revision 2-A of the report), NRC licensees may apply the PWROG acceptance criteria methodology and data requirements as defined in the subject TR (subject to the limitations and conditions when referencing the subject TR as documented in this SE) which may be used to satisfy the NEI-03-08 "Needed" criterion referenced in Section 7.5 of the MRP-227, Revision 1-A, but only in a limited case. Specifically, the PWROG's acceptance criteria and data requirements for the applicable component ID item assessment in the subject TR may only be applied if the I&E criteria for the component type in MRP-227, Revision 1-A remain applicable to the component when considering 80-year aging considerations in a gap analysis or as defined in



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MRP-227, Revision 2. For RVI components whose I&E criteria would change under 80-year aging considerations from those for the same components in MRP-227, Revision 1-A, the applicable ID item assessments for the components in the subject TR would not apply to the 80-year aging considerations for the components. Therefore, the limitation identified in this section needs to be addressed in subsequent individual plant licensing applications when referencing the subject TR.

### 3.3 Bases for Closing Out Previously Issued Conditions from NRC's Safety Evaluation for Topical Report WCAP-17096-A, Revision 2

In the NRC's SE dated May 3, 2016, for WCAP-17096-NP, Revision 2, several limitations and conditions were issued based on the PWROG's previous acceptance criteria and data analysis bases for component RVI locations, as defined in Appendix A, C, or E. In Section 2.2 of the subject TR, the PWROG amended the WCAP-17096 methodology for the purpose of resolving those NRC limitations and conditions identified in NRC's SE for WCAP-17096-NP, Revision 2.

The following subsections provide a discussion of the NRC staff's review of the PWROG's bases for resolving and closing out these conditions as detailed in the subject TR. As discussed below, the NRC staff considers all conditions from NRC's SE, dated May 3, 2016, to have been met (relative to the issuance of the conditions in the May 3, 2016 SE for WCAP-17096-NP, Revision 2) because: (1) the PWROG's basis for closing the stated condition in the subject TR, including supplemental information from the PWROG's RAI responses, was sufficient for closing the condition opened in the May 3, 2016 SE, or (2) the PWROG's basis for closing the stated condition in the subject TR, including supplemental information from the PWROG's RAI responses, was not sufficient to close out the condition identified in NRC's SE for WCAP-17096-NP, Revision 2, and the NRC staff proceeded to open a new condition or limitation in this SE for resolution of the previous condition.

#### 3.3.1 Basis for Closing Out Condition 1 for Group 1 RVI Components – Babcock and Wilcox Designs

The components that apply to Condition 1 (C-01) (WCAP-17096-NP, Revision 2) for Group 1 RVI components are identified below:

- core barrel cylinder and welds
- core former plates
- internal and external baffle-to-baffle (BB) bolts
- core barrel-to-former (CF) bolts
- locking devices for CF bolts and external BB bolts
- core baffle plates

The PWROG proposed to resolve C-01 (WCAP-17096-NP, Revision 2) for Group 1 RVI components by inclusion of the following additional action to the applicable section of Appendix A of the subject TR:

> *"The licensee shall submit the plant-specific analysis for this component item within one year after the inspection of the linked Primary item when the results trigger the expansion criteria in MRP-227."*

During its review, the NRC staff discovered two apparent inconsistencies with the PWROG proposed resolution of C-01 (WCAP-17096-NP, Revision 2) for the referenced Group 1 RVI components. Specifically, the NRC staff observed that the PWROG only modified Appendix A to include the



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additional actions for Group 1 RVI components related to EPRI MRP “Expansion” category components. However, the NRC staff determined that the C-01 (WCAP-17096-NP, Revision 2) for Group 1 RVI components also applies to the “Primary” core baffle plates, and the PWROG did not include an analogous type of additional actions in Appendix A to the subject TR, Section A.1.9 for the core baffle plates. Furthermore, NRC staff also verified that the EPRI MRP’s basis for dispositioning flaws in core baffle plates, or in any of the Group 1 RVI components “Expansion” category components also permitted disposition for Appendix A (specifically Sections A.2.5 to A.2.9) to be based on component repair or replacement activities as an alternative to a plant-specific analysis of the impacted component. The NRC staff addressed these apparent gaps in information through the issuance of RAI-02 and RAI-03 (see Reference 23).

In its response to RAI-02 (Reference 26), the PWROG amended the additional actions for the referenced Group 1 RVI components “Expansion” components in Section 2.2 to the subject TR and in Sections A.2.5 to A.2.9 of Appendix A to the subject TR to include component replacement or justified alternative disposition process options in the statement. In its response to RAI-03 (see Reference 26), the PWROG amended the subject TR Section 2.2 and the data analysis basis in the subject TR, Appendix A, Section A.1.9 for the “Primary” baffle plates to include the following additional action for the baffle plates:

> “…*the licensee is required to submit the LEFM [linear elastic fracture mechanics] analysis for the baffle plates under normal/upset condition loads, considering IASCC, within one year of the inspection of the Primary component item for NRC to determine whether review and approval are needed if the inspection results indicate aging not meeting the acceptance criteria in Table 5-1 of MRP-227.*”

The NRC staff finds that the PWROG’s incorporation of these additional actions in the subject TR is sufficient to close C-01 (WCAP-17096-NP, Revision 2) for Group 1 RVI components because the timing for submitting the plant-specific or LEFM analysis is appropriately incorporated into the subject TR and these additional actions now form the basis for submitting the plant-specific analysis to the NRC. Section 3.3.11 of this SE provides additional clarification on the use of these additional actions. Based on these considerations, C-01 (WCAP-17096-NP, Revision 2) for Group 1 RVI components has been met and is closed.

### 3.3.2 Basis for Closing Out Condition 1 for Group 2 RVI Components – Babcock and Wilcox Designs

The components that apply to C-01 (WCAP-17096-NP, Revision 2) for Group 2 RVI components are identified below:

- vent valve assembly, top and bottom retaining rings
- flow distributor (FD) bolts
- upper thermal shield (UTS) bolts, and their associated locking devices
- lower thermal shield (LTS) bolts, and their associated locking devices
- surveillance specimen holder tube (SSHT) bolts

The PWROG proposed to resolve C-01 (WCAP-17096-NP, Revision 2) for Group 2 RVI components by inclusion of the following additional actions to the applicable section in Appendix A to the subject TR:



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(1) the vent valve top and bottom retaining rings, FD bolts, and bolt locking devices:

*"The licensee shall submit the plant-specific analysis for this component within one year after the inspection if the results do not meet the acceptance criteria in MRP-227,"* or,

(2) the UTS bolts and bolt locking devices, LTS bolts and bolt locking devices, and the SSHT bolts and bolt locking devices:

*"The licensee shall submit the plant-specific analysis for this component within one year after the inspection of the linked Primary item when the results trigger the expansion criteria in MRP-227."*

The NRC staff finds that the PWROG's incorporation of these additional actions in the subject TR is sufficient to close C-01 (WCAP-17096-NP, Revision 2) for Group 2 RVI components because the timing for submitting the plant-specific analysis is appropriately incorporated into the subject TR and the additional actions now form the basis for submitting the plant-specific analysis to the NRC. Section 3.3.11 of this SE provides additional clarification on the use of these additional actions. Based on these considerations, C-01 (WCAP-17096-NP, Revision 2) for Group 2 RVI components has been met and is closed.

### 3.3.3 Basis for Closing Out Condition 1 for Group 3 RVI Components – Babcock and Wilcox Designs

The components that apply to C-01 (WCAP-17096-NP, Revision 2) for Group 3 RVI components are identified below:

- control rod guide tube (CRGT) assembly – CRGT spacer castings, and
- incore monitoring instrument (IMI) guide tube spiders and associated spider-to-lower grid rib section welds

The PWROG proposed to resolve C-01 (WCAP-17096-NP, Revision 2) for Group 3 RVI components by the inclusion of the following additional action to the applicable sections in the subject TR, Appendix A:

*"The licensee shall submit the plant-specific analysis for this component within one year after the inspection if the inspection results do not meet the acceptance criteria in MRP-227."*

The NRC staff finds that the PWROG's incorporation of the additional action to the subject TR is sufficient to closeout C-01 (WCAP-17096-NP, Revision 2) on the Group 3 RVI components because the basis for submitting the referenced analysis is appropriately incorporated in the subject TR and now forms the basis for submitting the plant-specific analysis to the NRC. Section 3.3.11 of this SE provides additional clarification on the use of this additional action. Based on these considerations, C-01 (WCAP-17096-NP, Revision 2) for Group 3 RVI components has been met and is closed.



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3.3.4 Basis for Closing Out Condition 2 for High Fluence Components – Babcock and Wilcox, Combustion Engineering, and Westinghouse Designs

As issued in NRC's SE, dated May 3, 2016, for WCAP-17096-NP, Revision 2, Condition 2 (C-02) (WCAP-17096-NP, Revision 2) pertains to high fluence RVI components that are projected to have neutron fluence exposures in excess of $3x10^{21}$ n/cm$^2$ (E > 1.0 MeV) at the end of the licensed service life of the plant or service period of interest. The condition addresses the crack growth rate (CGR) models for the high fluence stainless steel (SS) RVI components in the B&W, Combustion Engineering (CE) and W-design PWRs.

Section 2.2 of the subject TR includes the following additional action in the flaw tolerance section of the component ID item assessment for each of the specified C-02 (WCAP-17096-NP, Revision 2) RVI components that are included in either Appendix A, C, or E of the subject TR:

> "*One appropriate model is available in EPRI Report Models of Irradiation-Assisted Stress Corrosion Cracking of Austenitic SS in Light Water Reactor Environments, Volume 2: Disposition Curves Application [46], which provides the technical basis for ASME Code Case N-889 [52].*"

The NRC staff determined that additional clarification was needed to verify the basis for using the EPRI report on stress corrosion cracking published in 2014 (cited as Reference 46 in the subject TR) for component CGRs. Specifically, the PWROG indicated that the CGR models in the referenced EPRI report (References 28 and 29) were also used as the CGR criteria basis for the industry's development of ASME Code Case N-889. At the time of NRC's review of the subject TR, staff verified that the cited EPRI report had not yet been submitted for NRC review and that NRC's review of the ASME Code Case N-889 was still pending review under the current 10 CFR 50.55a rulemaking update. Therefore, NRC staff concluded that the EPRI report is not an NRC-approved methodology for establishing component CGRs. The NRC staff addressed the use of unapproved guidance with the issuance of RAI-04 (Reference 23).

By letter dated January 14, 2022 (Reference 24), the PWROG responded to RAI-04 and provided additional information on the technical justification for the CGR model calibration bases of SS weld components.[4] The NRC staff evaluated the PWROG's response to RAI-04 as part of the review of the proposed CGRs for SS base metal and weld components that are subject to the neutron fluence ranges shown in Table 2 below.

**Table 2 – Neutron Fluence Values**

| Fluence (f) Range n/cm$^2$ (E>1.0 MeV) | Dose (dpa) |
|---|---|
| $f \leq 5x10^{20}$ | < 0.75 |
| $5x10^{20} < f \leq 1.4x10^{22}$ | < 0.75 and ≤ 20 |
| $f > 1.4x10^{22}$ | > 20 |

The NRC staff verified that the PWROG's response to RAI-04 has placed a selective bounding limit on the CGR basis for the SS component for some of the PWROG-conditional statements regarding selection of the appropriate CGR. In addition, NRC staff reviewed the applicability of the referenced 2014 EPRI report based on a separate NRC review of ASME Code Case N-889 in NRC's Regulatory

[4] The NRC staff notes that the PWROG is applying the same bounding CGR bases to crack growth evaluations for stainless steel base metal components.



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Guide (RG) 1.147 (Reference 29). In alignment with previously established NRC positions in the SE for WCAP-17096-NP, Revision 2 and given that the NRC has endorsed ASME Code Case N-889 in RG 1.147, Revision 20, as subject to staff conditions on use of the code case, the NRC staff determined that the PWROG-established conditional criteria should be reflected in this Final SE with a condition for the selected CGRs. This condition is intended to ensure that licensees relying on the subject TR would be applying CGR bases of the referenced 2014 EPRI report in a manner that is consistent with the staff's conditions placed on use of ASME Code Case N-889, as established in Table 2 of RG 1.147, Revision 20. Therefore, NRC staff has defined a new Condition 1 (C-01) (WCAP-17096-NP, Revision 3) in Attachment A of this SE.

C-01 (WCAP-17096-NP, Revision 3):
For stainless steel metal and weld components in the specified neutron fluence exposure ranges as shown in Table 2 of Section 3.3.4 of this SE, the PWROG has indicated that the CGRs are based on the CGR criteria fluence ranges specified in ASME Code Case N-889, as established by the EPRI models in the EPRI Report and cited as *Reference 46* in the subject TR. The NRC has endorsed Code Case N-889 in Regulatory Guide 1.147, Revision 20, with conditions for the specified neutron fluence exposure ranges. Therefore, if the CGR models in *Reference 46* of the subject TR are followed by a licensee using this TR, then the CGR models are subject to the NRC conditions on Code Case N-889, as defined in Table 2 of Regulatory Guide 1.147, Revision 20.

As a result of the PWROG's response to RAI-04, and its inclusion of the additional action in the subject TR, the NRC staff finds that the item addressed in RAI-04 is resolved and C-02 (WCAP-17096-NP, Revision 2) for High Fluence components for B&W, CE, and W is closed, as subject to the new C-01 (WCAP-17096-NP, Revision 3) for the applicable CGR basis.

### 3.3.5 Basis for Closing Out Condition 3 for Group 1 RVI Components – Combustion Engineering and Westinghouse Designs

The components that apply to Condition 3 (C-03) (WCAP-17096-NP, Revision 2) for Group 1 RVI components are identified below:

- CE-design welded core shroud assemblies
- CE-design bolted core shroud assemblies
- W-design baffle edge bolts
- W-design baffle-former assemblies

The PWROG proposed to resolve C-03 (WCAP-17096-NP, Revision 2) for Group 1 RVI components by inclusion of the following additional action to the applicable sections in the subject TR, Appendices C and E:

> *"The licensee shall submit the plant-specific or generic FMEA [failure modes and effects analysis] and confirmation of applicability of generic FMEA analysis within one year after any inspection that detects relevant conditions as defined in Table 5-2 (or Table 5-3) of MRP-227."*

The NRC staff finds that the PWROG's incorporation of the additional action in the subject TR, Appendices C and E, are sufficient to close C-03 (WCAP-17096-NP, Revision 2) for Group 1 RVI components because the basis for submitting the referenced analysis is appropriately incorporated in



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the subject TR and now forms the basis for submitting the plant-specific analysis to the NRC. Section 3.3.11 of this SE provides additional clarification on the use of this additional action. Based on these considerations, C-03 (WCAP-17096-NP, Revision 2) for Group 1 RVI components has been met and is closed.

3.3.6 Basis for Closing Out Condition 3 for Group 2 RVI Components – Combustion Engineering and Westinghouse Designs

The components that apply to C-03 (WCAP-17096-NP, Revision 2) for Group 2 RVI components are identified below:

- CE-design core support column bolts in CE designs with bolted core shroud (CS) assemblies[5]
- CE-design core support columns
- W-design lower support column bolts
- W-design cast and non-cast lower support column bodies

The PWROG proposed to resolve C-03 (WCAP-17096-NP, Revision 2) for Group 2 components by inclusion of the following additional actions in the applicable sections of the subject TR for the components identified below:

(1) for the bolted CE-design core support column bolts and the W-design lower support column bolts:

*"Note that the licensee shall submit the plant-specific analysis for the acceptable distribution of intact components within one year after any inspection that triggers the expansion criteria as defined in Table 5-2 (or Table 5-3) of MRP-227."*

(2) for the CE-design core support columns and the W-design cast or non-cast lower support column bodies:

*"Note that the licensee shall submit the plant-specific analysis for the acceptable distribution of intact components within one year after any inspection that detects the relevant conditions as defined in Table 5-2 (or Table 5-3) of MRP-227."*

(3) for the cast lower support column bodies:

*"If plant-specific functional analyses have been performed to respond to applicant/licensee Action Item 7 from MRP-227-A prior to the initial inspection of the components, licensees may compare the previously performed functionality analyses and need not resubmit plant-specific analyses unless the inspection findings are not bounded by the original analysis.*"

The NRC staff finds that the PWROG's incorporation of these additional actions in the subject TR is sufficient to closeout C-03 (WCAP-17096-NP, Revision 2) for the Group 2 RVI components because

[5] The CE-ID assessment for core support column bolts in CE-ID Item 1.1 of Appendix C of the Subject TR was deleted in the PWROG's response to RAI No. 7-(a). Thus, the staff acknowledges that C-03 (Revision 2 of TR) for Group 2 components no longer applies to Group 2 component types (including core support column bolts) in the two U.S. CE-design PWRs with bolted core shroud assembly designs. These PWRs (i.e., the PWR units at Palisades and Ft. Calhoun Stations) have entered permanently defueled, non-power operations per their amended DPR-20 and DPR-40 operating license and technical specification requirements.



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the basis for submitting the plant-specific analysis is appropriately incorporated in the subject TR and the additional actions now form the basis for submitting the plant-specific analysis to the NRC. Section 3.3.11 of this SE provides additional clarification on the use of these additional actions. Based on these considerations, C-03 (WCAP-17096-NP, Revision 2) for Group 2 RVI components has been met and is closed.

3.3.7 Basis for Closing Out Condition 3 for Group 3 RVI Components Items A and B – Combustion Engineering and Westinghouse Designs

The components that apply to C-03 (WCAP-17096-NP, Revision 2) for Group 3 RVI components are identified below:

- CE-design core support barrel assembly (CSB) upper flange weld
- CE-design CSB middle girth weld (MGW)
- CE-design CSB flexure weld
- CE-design CSB lower girth weld (LGW)
- CE-design CSB upper girth weld (UGW)
- CE-design CSB upper axial welds (UAWs)
- CE-design lower core support beams
- CE-design CSB middle axial welds (MAWs)
- CE-design CSB lower axial welds (LAWs)
- W-design core barrel assembly (CB) upper flange weld (UFW)
- W-design CB LGW
- W-design CB UGW
- W-design CB UAWs
- W-design CB lower flange weld (LFW)
- W-design CB MAWs
- W-design CB LAWs

In Section 4.0 of the NRC's SE, dated May 3, 2016, for WCAP-17096-NP, Revision 2, the NRC identified several conditions to be implemented when referencing Revision 2 of the TR. In the subject TR, the PWROG provided resolutions for how each condition is addressed. With respect to C-03 (WCAP-17096-NP, Revision 2) for Group 3 RVI components, Item a, the NRC staff established its position for this case (in Section 3.2.2.2.2 of NRC's 2016 SE) where a follow-up inspection for depth sizing of the detected crack in the referenced weld type is only necessary when the visual inspection of the specified weld reveals the presence of a surface-breaking flaw or crack. The NRC staff also established its position that the follow-up inspection of the weld should be performed at the next refueling outage (RFO) after the discovery of the crack.

To address the technical justification for a component re-inspection interval associated with C-03 (WCAP-17096-NP, Revision 2), Group 3 RVI components, Item a, the PWROG incorporated the following additional action to the applicable sections of the subject TR, Appendices C and E:

> *"In order to apply the TR methodology for cracks detected via one-sided visual examinations, if supplementary examinations to determine crack depth were not performed, a follow up examination must be performed no later than the next RFO to confirm the assumed crack growth rates are conservative, unless a technical justification for a longer verification interval is submitted."*



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The NRC staff finds that with the exception of the issue regarding neutron fluence values discussed below, the PWROG's incorporation of this additional action in the subject TR is sufficient to close C-03 (WCAP-17096-NP, Revision 2) for Group 3 RVI components, Item a, because: (1) the need for performing a follow-up inspection of the applicable components is consistent with the basis of C-03 (WCAP-17096-NP, Revision 2) and (2) the additional action is appropriately incorporated into the subject TR which now forms the basis for submitting the plant-specific analysis to the NRC. Section 3.3.11 of this SE provides additional clarification on the use of this additional action.

The NRC staff confirmed that the PWROG's reference in the subject TR to the 2014 EPRI report on stress corrosion cracking (see Reference 27) provides the basis for closing out C-03 (WCAP-17096-NP, Revision 2), Group 3 RVI components, Item a, by establishing the CGRs of the referenced SS base metal or weld components. The NRC staff also verified that the methodology in the referenced EPRI report has not been approved by the NRC staff. However, the NRC staff was able to review the applicability of the referenced 2014 EPRI report based on a separate NRC review conducted on code case N-889 in NRC's Regulatory Guide 1.147 (Reference 29). The NRC approved the code case with conditions as previously discussed and evaluated in Section 3.3.4 of this SE. Based on these considerations, the NRC staff evaluated the PWROG's response to RAI-04 as part of the review of the proposed CGRs for SS base metal and weld components that are subject to the neutron fluence ranges as described in Table 2 in Section 3.3.4 of this SE.

The NRC staff finds that the PWROG's inclusion of the additional actions in the subject TR Appendices C and E, and NRC staff's issuance of the new C-01 (WCAP-17096-NP, Revision 3)[6] on CGRs for PWR RVI components with neutron fluence exposures and ranges listed in this section are sufficient to close out C-03 (WCAP-17096-NP, Revision 2), Group 3 RVI components, Item a because the basis for submitting the referenced analysis is appropriately incorporated in the subject TR. Based on these criteria and assessments including the issuance of the new C-01 (WCAP-17096-NP, Revision 3), C-03 (WCAP-17096-NP, Revision 2), Group 3 RVI components, Item a, has been met and is closed.

With respect to C-03 (WCAP-17096-NP, Revision 2), Group 3 RVI components, Item b, the NRC staff established its position from the previous approach applied in WCAP-17096-NP, Revision 2, for the referenced Group 3 "Primary" components requiring performance of a plant-specific analysis for the components, with the basis for initiating the analysis being dependent on the EPRI MRP category for the component and whether or not the applicable Section 5 of Table 5-2 in MRP-227, Revision 1-A established any defined expansion criteria for the components. The PWROG addressed C-03 (WCAP-17096-NP, Revision 2), Group 3 RVI components, Item b, with the inclusion of the following additional actions for the applicable sections of the subject TR:

(1) CE-design "Primary" category CSB UFWs and MGWs and the W-design "Primary" category CB UFWs and LGWs

*"The licensee shall submit the plant-specific crack growth and fracture mechanics analysis justifying the detected flaws, including a justification of the allowable operating period before re-inspection, within one year after any inspection that triggers the expansion criteria as defined in Table 5-2 (or Table 5-3) of MRP-227."*

(2) CE-design "Primary" category CSB flexure welds and all the referenced Group 3 CE-and W-designs "Expansion" category components in CE-ID Items 5.1, 5.2, 5.3, 5.4, 6.1, and

[6] Condition C-01 is addressed in Section 3.3.4 in this SE.



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6.2 and W-ID Items 3.1, 3.2, 3.3, 4.2, and 4.3, where the components do not have applicable "Expansion" acceptance criteria in Table 5-2 or 5-3 of MRP-227, Revision 1-A:

"*The licensee shall submit the plant-specific crack growth and fracture mechanics analysis justifying the detected flaws, including a justification of the allowable operating period before re-inspection, within one year after any inspection that detects relevant conditions as defined in Table 5-2 (or Table 5-3) of MRP-227.*"

The NRC staff finds that the PWROG's incorporation of the additional actions in the subject TR is sufficient to closeout C-03 (WCAP-17096-NP, Revision 2) for the Group 3 RVI components, Item b, because the timing for submitting the plant-specific analysis is appropriately incorporated into the subject TR and the additional actions now form the basis for submitting the plant-specific analysis to the NRC. Section 3.3.11 of this SE provides additional clarification on the use of this additional action. Based on these considerations, C-03 (WCAP-17096-NP, Revision 2) for Group 3 RVI components, Item b, has been met and is closed.

3.3.8 Basis for Closing Out Condition 3 for Group 4 RVI Components – Combustion Engineering Designs

The components that apply to C-03 (WCAP-17096-NP, Revision 2) for Group 4 RVI components are identified below:

- CE-design core shroud plate-to-former plate welds
- CE-design core shroud plates
- CE-design core shroud remaining axial welds (CE-design units with welded core shrouds assembled in two vertical sections)
- CE-design core shroud remaining axial welds (CE-design units with welded core shrouds assembled with full height shroud plates)
- CE-design core shroud rib and rings

In Section 4.0 of NRC's SE for WCAP-17096-NP, Revision 2, NRC identified several conditions to be implemented when referencing Revision 2. In the subject TR, the PWROG provided resolution for how each condition is addressed. The PWROG resolved C-03 (WCAP-17096-NP, Revision 2) for Group 4 with the addition of the following additional actions to the applicable sections in the subject TR, Appendix C:

- CE-design "Primary" category core shroud plate-to-former plate welds and core shroud plates:

  "*The licensee shall submit the plant-specific fracture mechanics analysis justifying the detected flaws within one year after any inspection that triggers the expansion criteria as defined in Table 5-2 of MRP-227.*"

- CE-design "Expansion" category core shroud remaining axial welds in CE-ID Items 2.1 and 3.1 that do not have applicable "Expansion" acceptance criteria in Table 5-2 of MRP-227, Revision 1-A:

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



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With the exception of one item discussed below that needs further clarification regarding the PWROGs basis for the core shroud ribs and rings in CE-ID Item 3.2, the NRC staff finds that the PWROG's incorporation of these additional actions in the subject TR, is sufficient to closeout C-03 (WCAP-17096-NP, Revision 2) for the Group 4 RVI components because the basis for submitting the plant-specific analysis is appropriately incorporated in the subject TR and the additional actions now form the basis for submitting the plant-specific analysis to the NRC. Section 3.3.11 of this SE provides additional clarification on the use of these additional actions.

The NRC staff observed that the PWROG did not apply the second of these additional actions to the subject TR to the assessment for the core shroud ribs and rings in CE-ID Item 3.2 (applicable to CE-design PWRs with welded core shrouds using two vertical sections) based on the additions shown below:

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

The NRC staff also observed that, while the EPRI MRP clearly identified that the CSB ribs and rings are inaccessible (as stated in CE-ID Item 3.2 of MRP-227, Revision 1-A, Table 4-5), the EPRI MRP also identified that the NRC licensee needs to justify further service of the ribs and rings "by evaluation or by replacement" if *expansion* to the rib and ring components is triggered by the results of "Primary" inspections being applied to the core shroud plates in the units. The NRC staff addressed this gap in the information with the issuance of RAI-05 (Reference 23).

In the PWROG's response to RAI-05 (Reference 25), revisions were made to Section 2.2 to the subject TR to indicate that a functionality analysis, if triggered, is applicable to the core shroud ribs and rings in CE-design plants that have welded core shrouds using full height shroud plates. The PWROG also amended the "*Additional Action(s)*" in the applicable sections in Appendix C to the subject TR, CE-ID Item 3.2 to include the following additional action for the data analysis criteria that apply to the core shroud ribs and rings:

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

The NRC staff finds that the PWROG's incorporation of these additional actions in the subject TR, is sufficient to closeout C-03 (WCAP-17096-NP, Revision 2) for the Group 4 RVI components because the basis for submitting the plant-specific analysis is appropriately incorporated in the subject TR and the additional actions now form the basis for submitting the plant-specific analysis to the NRC. Section 3.3.11 of this SE provides additional clarification on the use of these additional actions. Based on all the considerations discussed in this section, C-03 (WCAP-17096-NP, Revision 2) for Group 4 RVI components has been met and is closed.



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3.3.9 Basis for Closing Out Condition 3 for Group 5 RVI Components – Westinghouse Designs

The components that apply to C-03 (WCAP-17096-NP, Revision 2) for Group 5 RVI components are identified below:

- Control rod guide tube lower flange welds
- Thermal shield flexures only for W-designed PWRs that include internal thermal shield assemblies
- CRGT LFWs in the remaining (i.e., accessible non-peripheral) CRGT assemblies

In Section 4.0 of NRC's SE for Revision 2 of the TR, NRC identified several conditions to be implemented when referencing Revision 2. In the subject TR, the PWROG provided resolutions for how each condition is addressed. The PWROG resolved C-03 for Group 5 by inclusion of the following additional actions to the applicable sections of the subject TR for the components identified below:

- W-design peripheral CRGT LFWs and the "Expansion" category non-peripheral CRGT LFWs:

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

- W-design thermal shield flexures:

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

The NRC staff finds that the PWROG's incorporation of these additional actions in the subject TR, is sufficient to closeout C-03 (WCAP-17096-NP, Revision 2) for the Group 5 RVI components because the basis for submitting the plant-specific analysis is appropriately incorporated in the subject TR and the additional actions now form the basis for submitting the plant-specific analysis to the NRC. Section 3.3.11 of this SE provides additional clarification on the use of these additional actions. Based on these considerations, C-03 (WCAP-17096-NP, Revision 2) for Group 5 RVI components has been met and is closed.



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3.3.10 Basis for Closing Out Condition 3 for Group 6 RVI Components – Combustion Engineering and Westinghouse Designs

The components that apply to C-03 (WCAP-17096-NP, Revision 2) for Group 6 RVI components are identified below:

- CE-design core support plates in the lower support structure
- CE-design fuel alignment plates in the upper internals assembly
- W-design lower support column forging or casting in the lower internals assembly
- W-design upper core plate in the upper internals assembly

In Section 4.0 of NRC's SE for WCAP-17096-NP, Revision 2, the NRC staff identified several conditions to be implemented when referencing Revision 2. In the subject TR, the PWROG provided resolutions for how each condition is addressed. The PWROG addressed C-03 for Group 6 by inclusion of the following additional action to the applicable sections of the subject TR for the components referenced CE-ID and W-ID items:

> *"The licensee shall submit the plant-specific analysis for this component within one year after any inspection that detects relevant conditions as defined in Table 5-2 (or Table 5-3) of MRP-227."*

The NRC staff finds that the PWROG's incorporation of this additional action in the subject TR, is sufficient to closeout C-03 (WCAP-17096-NP, Revision 2) for the Group 6 RVI components because the basis for submitting the plant-specific analysis is appropriately incorporated in the subject TR and the additional action now forms the basis for submitting the plant-specific analysis to the NRC. Section 3.3.11 of this SE provides additional clarification on the use of these additional actions. Based on these considerations, C-03 (WCAP-17096-NP, Revision 2) for Group 6 RVI components has been met and is closed.

3.3.11 Additional Clarification on Supplemental Analysis for Component ID Item Assessments in Topical Report WCAP-17096, Revision 3

Some of the PWROG component ID item assessments in the subject TR Appendices A, C, and E included additional actions for applicable components that may instruct NRC licensees to submit a supplemental analysis (when triggered) of the components that are within the scope of a given component ID item assessment for potential NRC review. In many cases, the additional actions for applicable components in the subject TR may specify performance of a specific type of analysis for the specified components. Accordingly, the need for performing these supplemental activities fall within the scope of the NRC licensee's own corrective action program, and a licensee's decision to determine the proper corrective action resolution that would be used under its 10 CFR Part 50, Appendix B program or NEI 03-08 industry initiative process could result in augmented inspections that are performed in accordance with the methodology in MRP-227, Revision 1-A.

The NRC staff acknowledges that the specific type of analysis that may be referenced in an additional action for an applicable component in the subject TR constitutes only one way of dispositioning levels of age-related degradation that exceed the levels for initiating corrective actions per the criteria that are defined for the applicable component type in MRP-227, Revision 1-A. However, consistent with the implementation criteria in Chapter 7 of MRP-227, Revision 1-A, the licensee has the option of performing and using any type of engineering analysis that differs from the analysis specified in the



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subject TR. The licensee may also justify further service of the components with proposed component repair or replacement activities and repair/replacement schedules that would ensure the intended functions of the components.

## 3.4 Clarification of Items Addressed in Topical Report WCAP-17096, Revision 3

### 3.4.1 Statements in MRP-227, Revision 2 Outside the Scope of Topical Report WCAP-17096, Revision 3

In Section 1 of the subject TR, the PWROG included the following statement:

> "*the MRP-227 inspection and evaluation guidelines are in development for SLR (80-year operating license), and can be expected to continue to evolve based upon lessons learned from operating experience. Therefore, each utility must remain cognizant of changes made…*"

The NRC staff notes that it has not yet approved the MRP-227 report for an 80-year operating license (e.g., MRP-227, Revision 2, or equivalent report submitted by the EPRI MRP for NRC approval).[7] Since the NRC has not yet approved MRP-227, Revision 2, for implementation by NRC licensees, the NRC staff finds these statements to be outside the scope of this SE.

### 3.4.2 Time Dependency Considerations for Topical Report WCAP-17096, Revision 3

In Section 1 of the subject TR, the PWROG clarified that: (1) the RVI AMP applies to both initial and subsequent license renewal (LR/SLR) periods, and (2) the analytical methods defined in the subject TR may be "*independent of time*" or "consider time dependency when appropriate." The NRC staff determined that some additional explanations are needed for these statements in the subject TR.

Specifically, the methodology in the subject TR accounts for the possibility that an NRC licensee could be performing an RVI component evaluation that is based on a time-dependent parameter defined by the current operating period (Refer to Criterion 3 in 10 CFR 54.3(a) [§54.3(a)]), and that the analysis may conform to all six of the criteria for defining a TLAA in §54.3(a). For those analyses that are determined to conform to the six criteria for TLAAs in §54.3(a), it is the NRC licensee's responsibility to ensure that: (1) these analyses are identified and evaluated as TLAAs in the LRA/SLRA in accordance with 10 CFR 54.21(c)(1) [§54.21(c)(1)], and (2) that the TLAAs are appropriately dispositioned in the LRA/SLRA in accordance with one or more of the acceptance criteria for TLAAs, as specified in §54.21(c)(1)(i), (ii) or (iii). For any RVI specific analysis that is determined to conform to the definition of a TLAA, the LRA/SLRA would also be required to include a Final Safety Analysis Report (FSAR)/Updated Final Safety Analysis Report (UFSAR) supplemental summary description for the TLAA in the LRA/SLRA, as required by 10 CFR 54.21(d) [§54.21(d)]. The NRC staff determined that no further review is necessary for the objectives of this SE since the need for identifying and evaluating TLAAs in LRAs/SLRAs is driven and set by the TLAA criteria specified in §54.3a, and the requirements for TLAAs in §54.21(c)(1) and §54.21(d).

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[7] The staff notes that EPRI MRP submitted an eighty-year version of MRP-227 (as MRP-227, Revision 2) for staff approval on May 9, 2022 (ML22129A140 for the EPRI MRP transmittal letter; ML22129A141 for the MRP-227, Revision 2 report) and the staff accepted MRP-227, Revision 2 for a formal review in a staff-issued email to EPRI MRP dated June 17, 2022 (ML22145A401). The staff's review of MRP-227, Revision 2 is ongoing and the staff's decision on MRP-227, Revision 2 is pending.



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3.4.3 ASME Code Criteria Applicability to Topical Report WCAP-17096, Revision 3

In Section 2 of the subject TR, the PWROG provided the following statements clarifying that specific acceptance standards in ASME Boiler and Pressure Vessel Code, Section XI Article IWB-3500 may not apply to the evaluation of PWR RVI component inspection results:

> "*While these acceptance standards are appropriate for the ASME Code Section XI inspections, including those that are specifically highlighted in MRP-227 as existing reactor internals inspection requirements, they are not fully applicable to the MRP-227 inspection recommendations. In particular, the linear flaw standards of IWB-3510, as intended to guard against propagation of cracks through the reactor pressure vessel, are not meaningful when applied to the removable internals components that do not provide a pressure boundary.*"

The NRC staff determined that the PWROG's general statements in this regard would not be valid for NRC licensees implementing MRP-based PWR RVI AMPs that: (1) rely on MRP-227, Revision 1-A, and (2) include Existing Program criteria for specified ASME Code Class 1 core support structure components or reactor internal interior attachments. Specifically, such criteria would apply to W- and CE-design RVI components that: (1) are defined as "Existing Program" components in Table 4-8 or 4-9 of MRP-227, Revision 1-A, and (2) are referenced in ASME Code Class inspection criteria by the line item for the specific component type in either Table 4-8 or 4-9 of MRP-227, Revision 1-A.[8] Although the NRC staff agrees with the PWROG conclusion that PWR RVI components are not defined by ASME International as ASME Code Class 1 reactor coolant pressure boundary components, they may be defined in the CLB as ASME Code Class 1 interior attachments or core support structure components under ASME Section XI, Table IWB-2500-1, Examination Category B-N-2 or B-N-3 criteria.

In addition, the EPRI MRP makes the following statements in the executive summary of MRP-227, Revision 1-A regarding 10 CFR 50.55a and ASME Code compliance:

> *"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."*

Therefore, for PWR RVI components that are defined in the CLB as ASME Code Class 1 Examination Category B-N-2 interior attachments or B-N-3 core support structure components (regardless of the MRP-227, Revision 1-A category classification), the acceptance criteria for the components are established according to one of the following In-Service Inspection (ISI) program bases:

- The applicable acceptance criteria for the component in ASME Section XI, Article IWB--3000

[8] These criteria do not apply to B&W designed PWRs as the EPRI MRP does not include any RVI components in the "Existing Program" category of MRP-227, Revision 1-A for B&W designed plants.



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- Through use of an NRC-approved ASME Section XI Code Case (Reference 30) endorsed in the most recent staff-issued version of RG 1.147 (Reference 29), as referenced in 10 CFR 50.55a(b) and subject to any limitations or conditions placed on use of the Code Case in the RG

- Through implementation of alternate, NRC-approved ISI acceptance criteria and data requirements that are authorized by NRC through implementation of a 10 CFR 50.55a(z) relief request review and approval process

The need to follow the above referenced ASME Section XI or 10 CFR 50.55a process criteria for RVI components that are included as ASME Code Class 1 or Class A components in the plant design is independent of the EPRI MRP category assigned to the component in MRP-227, Revision 1-A or in supplemental EPRI methodologies (e.g., EPRI MRP 2018-022).

3.4.4 Reinspection Frequencies for Baffle-Former Bolt Inspections

In Section 2.3 of the subject TR, the PWROG referenced the EPRI MRP's interim inspection criteria for W-design baffle-former bolt (BFB) inspections, as established in MRP-2016-021 (Reference 32) or PWROG-17071-NP, Revision 0 (Reference 34). The PWROG also referenced EPRI's MRP responses to the NRC's RAIs on the Interim Guidance Reports, dated July 13, 2017 (Reference 46), and NRC's approval basis for accepting NEI 03-08 versions (Reference 36).

In Section 2.3.2 of the topical report PWROG-17071-NP, Revision 0, the PWROG included the following "Needed" criterion, which was created based on a Westinghouse PWR licensee's proposed reinspection interval bases for its BFBs:

> *"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 NRC staff previously accounted for the plant-specific evaluation and reinspection need bases of W-design BFBs through the staff's issuance of A/LAI-01 in the NRC's SE dated April 25, 2019, for MRP-227, Revision 1-A. To address the A/LAI-01 in Revision 3 to the TR, the PWROG updated Appendix E to the subject TR, W-ID Item 6 to include the following additional actions for BFBs in the "*Additional Action(s)*" section of the component ID item:

> "*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.*"

The NRC staff finds that the supplemental BFB analysis and reinspection criteria is acceptable because they are consistent with the NRC's recommendations for W-design BFB reinspection intervals, as discussed in the NRC's April 25, 2019, SE for MRP-227, Revision 1-A (Reference 17).



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For LRA/SLRA of W-design PWRs that include the subject TR within the scope of its LRA/SLRA "PWR Vessel Internals" AMPs, licensees may address the A/LAI-01-based reinspection intervals for the BFBs in either the "*monitoring and trending*" or "*operating experience*" program element assessments of their AMPs by referencing the PWROG's statements in the "*Additional Action(s)*" section of Appendix E of the subject TR, W-ID Item 6. The timeframe for determining whether a supplemental analysis for BFB is necessary is appropriately defined in the additional action for the components in the subject TR. Section 3.3.11 of this SE provides additional clarification regarding the use of corrective actions for a given component and have been incorporated in the subject TR. Based on these considerations, the NRC staff finds the methodology guidance in Section 2.3 of the subject TR to be acceptable.

### 3.4.5 Analyses for Ultrasonic Test Results of Bolted Assemblies

In Section 3.3 of the subject TR, the PWROG discussed its bases for evaluating the results of ultrasonic test (UT) inspections performed on bolted RVI assemblies, including, but not limited to the evaluation of BFBs used in the fabrication of the W-design baffle-former assemblies. Section 3.3 of the subject TR also includes discussions of acceptable bolting pattern analysis (ABPA) methodologies and use of acceptable bolting patterns (ABPs) that may be used to justify the failure of a prescribed number of bolts without causing structural integrity issues with the RVI assembly containing the bolts or to justify reinspection intervals of the bolts contained in these assemblies. Specifically, the PWROG added the following additional action to the applicable sections of the subject TR to include ABPAs for bolted assembly evaluations:

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

The NRC staff transmitted RAI-07, Parts (a), (b), and (c) to the PWROG on September 21, 2020 (Reference 23), to address three items which were related to the PWROG's basis for using the ABPA in the evaluation of degraded bolts in RVI bolted connections. In the PWROG's response to RAI-07, Parts (a), (b), and (c) dated January 14, 2022 (Reference 43), PWROG's response to Part (a) discusses the ABPA from the proprietary version of WCAP15029P-A (Reference 37) for CE-design core shroud bolts and barrel-to-shroud bolts; the only two CE-designed power plants that link the inspection and ABPA evaluation criteria are the nuclear reactors at the Fort Calhoun and Palisades nuclear power stations. The PWROG indicated that the licensees for the PWRs at Fort Calhoun Station and Palisades Station have permanently ceased operations of the units. The current version of Operating License No. DPR-20 for Palisades Station indicates that the site is now operating in accordance with permanently defueled plant conditions and technical specification requirements. Similarly, the current version of Operating License No. DPR-40 for Fort Calhoun Station indicates that the site is now operating in accordance with permanently defueled plant conditions and technical specification requirements.



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The PWROG addressed this RAI by amending Appendix C to the subject TR by deleting the component acceptance criteria methodology and data requirements assessments for core shroud bolts and barrel-to-shroud bolts in Appendix C to the subject TR, CE- Item ID 1 and 1.2, along with the corresponding criteria for the linked “Expansion” category core support column bolts in Appendix C to the subject TR, CE-ID Item 1.1 and associated flow charts for these components in the corresponding component ID item in Appendix D to the subject TR. Given that the NRC staff confirmed that the referenced ABPA acceptance criteria methodology and data requirements are only for the specified bolting types in non-operating CE-design reactors (i.e., Fort Calhoun and Palisades), the NRC staff finds these proposed changes in the subject TR for CE-ID item to be acceptable, as the MRP-227-based AMPs would no longer be implemented at these sites because they have both permanently ceased power operations. Therefore, RAI-07, Part (a) is resolved.

Additionally, in the PWROG’s January 14, 2022, response to RAI-07, Part (b), the PWROG confirmed that the methods in WCAP-18034-P are used to supplement acceptance criteria methodology and data requirements in WCAP-15029-P-A, Revision 1, for application to the W-ID item assessments for the BFB in Appendix E to the subject TR, W-ID Item 6 and for the barrel-to-former bolts in Appendix E to the subject TR, W-ID Item 6.1. The NRC staff confirmed that the PWROG amended Appendix E to the subject TR, W-ID Items 6 and 6.1 to include these clarifications. Based on the PWROG’s clarifications and the amendment of Appendix E to the subject TR, W-ID Items 6 and 6.1, the NRC staff finds that the subject TR appropriately addresses which NRC-approved ABPA methodologies that will be applied to the assessments of W -design BFBs and core barrel-to-former bolts. Therefore, RAI-07, Part (b) is resolved.

Moreover, in the PWROG’s January 14, 2022, response to RAI-07, Part (c), the PWROG restated its position that there are no NEI 03-08 “Mandatory,” “Needed” or “Good Practice” criteria in the subject TR, but indicated that the subject TR can be used to meet the NEI 03-08 “Needed” criterion as specified in Section 7.5 of MRP-227, Revision 1-A. Therefore, the NRC staff finds this aspect of the response to RAI-07, Part (c) acceptable because it is consistent with the implementation bases in Section 7 of the MRP-227, Revision 1-A.

The PWROG also revised its response to RAI-07, Part (c) in the PWROG’s letter of January 14, 2022 (Reference 24 for the transmittal letter; Reference 25 for the revised RAI response), that superseded the PWROG’s prior response to RAI-07, Part (c) previously submitted in the PWROG’s letter dated March 30, 2021 (see Reference 43). Specifically, the PWROG indicated that the discussion in Section 3.3 of the subject TR regarding the confirmation of the plant-specific applicability of a generic ABPA is a process step that is discussed within the scope of the W-ID Item 6 methodology in the subject TR, rather than Section 3.3 of the subject TR. Therefore, the PWROG revised RAI-07, Part (c) in its response transmitted on February 8, 2022 (see Reference 43) as discussed below:

- Replacement of the “Approach” section statement of W-ID Item 6: “*generic work completed in previous PWROG program*” with the statement “*generic ABPAs that were previously performed should be confirmed to be applicable on a plant-specific basis*”

- Deletion of the following statement in Section 3.3 of the subject TR: “*The plant-specific applicability of the generic acceptable patterns must be confirmed by the user, since certain inputs may have been changed over two decades since these patterns were originally developed.*"

The NRC staff did not find the PWROG’s changes to the RAI responses to be acceptable (i.e., the complete deletion of the statement in Section 3.3 to the subject TR and limited scope on use of



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APBAs limited only to the bolting assessments for W design BFB in the subject TR, Appendix E, W-ID Item 6) because, even though changes were made to the subject TR in the PWROG's January 13, 2022, RAI response, the PWROG was still using NRC-approved APBAs, or have potential use of APBAs, in other ID items assessments in the subject TR Appendices A or E (including, but not limited to Appendices A or E):

- WCAP-17096-NP, Revision 3, Appendix E, W-ID Item 6.1 for W-design Expansion category barrel-to-former bolts (which, like W-ID Item 6 for W-design BFB, references use of the NRC-approved ABPA methodology in Westinghouse Class 2 Proprietary Report No. WCAP-15029-P-A, Revision 1 (Reference 38)).

- Multiple B&W-ID Item assessments in WCAP-17096-NP, Revision 3, Appendix A for bolted B&W-design Primary or Expansion category bolting types that referenced potential use of ABPAs for ensuring the integrity of the bolted assemblies (where the PWROG had indicated that ABPA methodology could be developed for the applicable bolting type).

With respect to this observation, the NRC staff has previously approved the use of ABPAs only for W-design baffle-to-former bolts and barrel-to-former bolts (WCAP-17096-NP, Revision 3, Appendix E, W-ID Item 6 and 6.1 components) based on NRC staff's past approval of WCAP-15029-P-A, Revision 1, in the NRC's SE (Reference 33). Additionally, in accordance with the basis in the NRC's SE WCAP-15029-P-A, Revision 1, the scope of the NRC approval was limited to only the deterministic ABPA methodology basis which would be used to generate an ABP or set of ABPs for the bolts. NRC's WCAP-15029-P-A, Revision 1, SE did not cover any methodology for the probabilistic Monte Carlo methods that might be used to generate a set of unit-specific bolting patterns (BPs) and that would then be compared against the NRC-approved ABP or set of ABPs for the W-design baffle-to-former bolts and barrel-to-former bolts. The NRC staff has not approved any ABPA and Monte Carlo methodologies for generating ABPs and unit-specific BPs for other types of W-bolted components or for CE-design or B&W-designed bolted component types. Therefore, with the inclusion of the new Condition 2 (new C-02) (WCAP-17096-NP, Revision 3) (discussed below in Section 3.4.6), if BPs in these other bolted component types are evaluated using unapproved ABPA, these evaluations should be made available to the NRC as discussed in Section 3.4.6 of this SE, which includes the NRC staff's evaluation of the PWROG's response to RAI-13, Part (b) (Reference 26) and the linked resolution of RAI-07, Part (c)**.**

### 3.4.6 Resolution of Operating Experience for Westinghouse-Design Baffle-Former Bolts, Thermal Shield Flexures, and Use of Predictive Bolting Pattern Models or Methods

In Section 2.3 of the subject TR, the PWROG discussed its basis for dispositioning relevant OE with cracking in W-design BFB. The NRC staff observed that due to the timing of the subject TR publication which took place in July of 2019, the PWROG did not address/discuss the BFB degradation that occurred at the Salem, Unit 1 facility during the reinspection for the spring 2019 outage, following the BFB replacement in the unit during the spring of 2016. Indications in the thermal shield flexures were also found during the spring of 2019 reinspection at Salem, Unit 1. WCAP-17096-NP, Revision 3, Appendix E, W-ID Item 9, provides evaluation methods for detected flaws in thermal shield flexures. The spring of 2019 reinspection at Salem, Unit 1 resulted in more than the expected number of degraded BFBs predicted by the methodology in WCAP-17096-NP, Revision 3, Appendix E, W-ID Item 6, which describes the basic framework for a probabilistic predictive model as a basis for evaluating BFB reinspection intervals. W-ID Item 6 states that the probabilistic model must



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predict both the proportion and spatial distribution (i.e., patterns) of failed bolts at certain time intervals. Acceptability of at least 95 percent of predictive model outputs relative to the acceptable bolting pattern is recommended as a basis for setting the reinspection interval. ABPA for “as-found” inspection results is based on the NRC-approved methodology in WCAP-15029-P-A, Revision 1.

On September 18, 2019, during an NRC closed meeting, Westinghouse (in partnership with the PWROG), gave a presentation (Reference 37) to NRC staff on “Westinghouse Baffle-Former Bolt Predictive Methodology”. The presentation provided an overview of the proprietary methodology for probabilistic predictive evaluations of BFB degradation as stated in the basic framework of WCAP-17096-NP, Revision 3, Appendix E, W-ID Item 6. The presentation included OE on the BFB degradation from the spring of 2019 outage at Salem, Unit 1. On September 21, 2020, the NRC staff issued RAI-13, Parts (a) to (g) (Reference 23) to address the unexpected number of degraded BFBs from the spring 2019 outage at Salem, Unit 1 (including information related to the thermal shield flexure indications) and clarifications provided in the September 2019 Westinghouse presentation on the probabilistic predictive methodology for evaluating BFB degradation.

The NRC staff reviewed the PWROG’s September 2019 Westinghouse presentation on BFB degradation OE that provided information on how the predictive, probabilistic evaluation outputs compare to BFB reinspection data. However, the NRC staff determined that it would need further verification that the predictive, probabilistic method for the BFB was suitable for setting reinspection intervals of the bolts based on generating predictive outputs starting from an initial degraded pattern. In the PWROG’s RAI-13 response (see Reference 39), the PWROG responded to RAI-13, Parts (a) – (g). In the PWROG’s response to RAI-13 Part (a), the PWROG explained that the validity of the Westinghouse predictive, probabilistic model is demonstrated by considering the results from previous applications, which were used to predict both initial BFB UT results and reinspection BFB UT results, as shown in the September 2019 Westinghouse presentation. The NRC staff noted from its review of the response to RAI-13 Part (a) that the reinspection BFB UT results were within the bounds of the predictive model, for all except for one plant. Based on considerations and the NRC staff’s review of the response to RAI-13, Part (a), the NRC staff finds that the predictive bolting pattern model is an acceptable method for establishing the reinspection interval for a specified BFB because the NRC staff has confirmed that the ABPA model is capable of generating a conservative, predictive set of future bolting pattern outputs starting from an initial degraded bolting pattern (as established from the last inspections performed on the assembly), and therefore, the issue in RAI-13, Part (a) is resolved.

With respect to the issue identified in RAI-13, Part (b), the NRC staff observed that if other vendors used probabilistic methods for bolted assemblies that are different from those described in the September 2019 Westinghouse presentation, these new probabilistic methods would need to be submitted for NRC review and approval. In the PWROG’s response to RAI-13, Part (b), the PWROG explained that WCAP-17096-NP, Revision 3, Appendix E, W-ID Item 6, specifies that each application of the predictive model methodology in W-ID Item 6 will be submitted to the NRC for information if the percentiles defined in MRP 2017-009 are exceeded, and that based on BWRVIP 2019-016 (Reference 47), the licensee submitting the request has the responsibility for making any proprietary details of the predictive methodology available to the NRC. Also, the PWROG stated that Framatome performs predictive BFB evaluations for W-design plants and that the methodology in Section A.1.8 of the subject TR is applied. The NRC staff finds that for W-design plants, the specification in WCAP-17096-NP, Revision 3, Appendix E, W-ID Item 6 to submit the plant-specific application of the predictive model methodology defined in W-ID Item 6 is an acceptable way for making the details of the predictive methodology available to the NRC. In accordance with WCAP-17096-NP, Revision 3, Appendix E, W-ID Item 6, submittal to the NRC is only for information and is only necessary if: (1) MRP 2017-009 indicates that the subsequent inspection interval is not to exceed 6 years, or (2) there is an evaluation to lengthen the determined inspection interval or to exceed the maximum inspection



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interval recommended in MRP 2017-009. The staff concludes that for W-design plants, the specification in WCAP-17096-NP, Revision 3, Appendix E, W-ID Item 6, to submit the plant-specific application of the predictive model methodology defined in W-ID Item 6 is an acceptable way for making the details of the predictive methodology available to the NRC.

Furthermore, the NRC staff observed that the PWROG's referenced use of Framatome ABPA and predictive methodologies for B&W-design bolted assembly types have yet to be developed by Framatome at the time of the development of this SE report. The NRC staff would expect that the plant-specific bolting evaluation methods[9] for the bolted assembly type to be placed into the NRC licensee's corrective actions process consistent with either the requirements in 10 CFR Part 50, Appendix B, Criterion XVI or else the NEI 03-08 "Needed" corrective action specified in Section 7.5 of the MRP-227, Revision 1-A. The staff also expects these evaluation methods to be fully defined, discussed, and justified in an applicable site-specific document or record. This is part of a new Condition (C-02) (WCAP-17096-NP, Revision 3) as defined and discussed in Table A-1 of Attachment A of this SE, which relates to NRC staff's evaluation items discussed in RAI-07 Part (c) (as carried over in Section 3.3.5 of this SE) and RAI-13 Part (b) as discussed and evaluated above. Based on the addition of C-02 (WCAP-17096-NP, Revision 3), the items discussed in RAI-07, Part (c) and RAI-13, Part (b) are resolved.

C-02 (WCAP-17096-NP, Revision 3):
For bolted assembly analysis methods that use predictive, probabilistic ABPA models, the models and methods should appropriately account for limits on the number of allowable bolts with presumed failed conditions, geometric bolt failure considerations (i.e., bolt clustering considerations), and bolting analysis reliability considerations (i.e., addressing 95% reliability confidence limits.) These models and methods should be appropriately documented and justified in a site-specific or owner-defined record.

With respect to the items discussed in RAI-13 Part (c), the NRC staff observed that potential errors in the UT data of the BFBs, in the inputs into the predictive methodology,or in the methodology itself could have caused the unexpected number of degraded BFBs discovered during the spring of 2019 outage at Salem, Unit 1. In the PWROG's response to RAI-13, Part (c), it discussed the root cause evaluation of the unexpected number of degraded BFBs from the spring of 2019 outage at Salem, Unit 1. Specifically, in its root cause evaluation, the PWROG concluded that the detected degradation in the thermal shield flexures resulted in increased applied stresses to the BFBs as a result of additional vibrational loads disseminating from the thermal shield, which lead to the unexpected number of degraded BFBs at Salem, Unit 1. Based on the conclusion of the root cause evaluation, the PWROG stated that Section 3 of the subject TR will be revised to clarify that the process of evaluating degradation (in general) in the subject TR assumes that each component degradation is independent of degradation of other components. The NRC staff reviewed the excerpts of the root cause evaluation provided in the supplement, and determined that the unexpected number of degraded BFBs in the spring 2019 outage at Salem, Unit 1 was not due to errors in the BFB UT data, the inputs into the predictive methodology, or in the methodology itself because the root cause of the degradation found in the BFB and the number of BFB found with relevant crack-like indications was

[9] This is in reference to both: (1) a deterministic ABPA methodology for generating an approved bolting pattern [ABP] or set of ABPs used for structural integrity acceptance criteria objectives of the bolted assembly, and (2) either a deterministic or probabilistic bolting pattern methodology for generating a set of unit specific bolting patterns (BPs) for the assembly that would then be compared to the ABP or set of ABPs.



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vibrations coming from the degraded thermal shield flexures. Therefore, the item discussed in RAI-13 Part (c) is resolved.

With respect to the item discussed in RAI-13 Part (d), the NRC staff observed that the 95 percent reliability criterion, i.e., a 95 percent probability of retaining an acceptable bolting pattern at reinspection compared against plant specific ABPAs, means that there is a 5 percent probability that the predictive model would result in an unacceptable bolting pattern at reinspection. In the PWROG's response to RAI-13 Part (d), it states that the 95 percent reliability criterion in terms of an acceptable bolting pattern adds a layer of conservatism to a scenario that is already judged to have low risk significance with respect to core damage frequency. The PWROG also stated that passive failures, such as BFB failures, are not typically modeled in plant-specific probabilistic risk assessments (PRAs) since failure probabilities associated with passive failures are several orders of magnitude lower than other active failures. The PWROG discussed the conservatisms in both the Westinghouse predictive model, and ABPA methodology (in WCAP-15029-P-A, Revision 1) that contribute to the low-risk significance of BFB degradation. The NRC staff especially observed the conservatism in what constitutes as core damage for a given BFB-related event and verified that this core damage criterion is more conservative than that used in PRAs as defined in the PRA Standard. The NRC staff confirmed that this referenced PRA Standard is the same standard for PRA evaluations specified in RG 1.174, Revision 3, for evaluating risk due to changes to the licensing bases. The NRC staff finds that the 95 percent reliability criterion is acceptable when the Westinghouse predictive model and ABPA methodologies are used for evaluating BFB degradation because of the conservatisms in both methodologies. However, the NRC staff determined that if the Framatome predictive model and ABPA methodologies are used for evaluating BFB degradation, the conservatisms in the methodologies would need to be confirmed for the acceptability of the 95 percent reliability criterion. This is addressed as part of the new C-02 (WCAP-17096-NP, Revision 3), as defined and discussed in Table A-1 of Attachment A in this SE. Based on the addition of new C-02 (WCAP-17096-NP, Revision 3), the item in RAI-13 Part (d), is resolved.

The methodology in WCAP-17096-NP, Revision 3, Appendix E, W-ID Item 6, refers to consideration of a portion of untestable bolts as functional when evaluating a bolting pattern in general or evaluating a bolting pattern with clustering. With respect to the item discussed in RAI-13 Part (e), the NRC staff observed that the subject TR does not include any specific guidance on how a plant should evaluate untestable bolts to determine the extent to which they may be credited as functional structural members. In the PWROG's response to RAI-13 Part (e), it stated that MRP-227, Revision 1-A allows a certain number of untestable bolts to be excluded when performing the MRP-227 inspections of BFBs, and that in general, when evaluating reduced bolting patterns based on inspection results, the most conservative approach is to consider all untestable bolts as failed. The PWROG also stated that a sampling approach was determined appropriate by the NRC, referring to the NRC's SE for WCAP-17096-NP, Revision 2 (Reference 13). The PWROG stated that the sampling approach is also intended to be used when untestable bolts are relevant to the determination of clustering in the reduced bolting pattern, and that Note 1 of Step 2 of WCAP-17096-NP, Revision 3, Appendix E, W-ID Item 6, will be revised to reflect this intent. The NRC staff reviewed the response and the NRC's SE of WCAP-17096-NP, Revision 2, regarding treatment of untestable bolts. The NRC staff finds that the methodology summarized by the PWROG for evaluating untestable bolts to determine the extent to which they may be credited as functional structural members acceptable because (1) treating all untestable bolts as failed (meaning they are not credited as a load bearing member) is conservative; and (2) adding the statistically determined number of failed bolts in the untested (i.e., uninspected) population to the number of failed bolts in the tested (i.e., inspected) population provides a conservative means of estimating the total number of failed bolts, as determined in the NRC SE that is included in Revision 2 of the TR. Additionally, consistent with the new C-02 (WCAP-17096-NP, Revision 3), use of predictive bolting pattern methodologies for PWR bolted assembly objectives



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should be appropriately defined, discussed, and justified in a site-specific document or record. Based on these considerations and the addition of the new C-02 (WCAP-17096-NP, Revision 3), the items discussed in RAI-13 Part (e), is resolved.

With respect to the item discussed in RAI-13, Part (f), the NRC staff observed that the language on reinspection intervals in WCAP-17096-NP, Revision 3, Appendix E, W-ID Item 6, is inconsistent with that in MRP-227, Revision 1-A and MRP 2017-009. In its response to RAI-13, Part (f), the PWROG acknowledged the existence of the perceived inconsistency and stated that step (c) of WCAP-17096-NP, Revision 3, Appendix E, W-ID Item 6, will be revised to include a statement that a submittal of a deviation to MRP-227 guidelines would be needed for reinspection intervals that are beyond 10-years. The NRC staff reviewed the response and finds that the additional statement on submittal of a deviation to MRP-227 guidelines to be acceptable because the PWROG's basis is consistent with EPRI MRP programmatic deviation identification and resolution criteria for these types of programs, as defined in Section 7 of MRP-227, Revision 1-A. The item discussed in RAI-13 Part (f), is resolved.

With respect to the item discussed in RAI-13, Part (g), the NRC staff observed that the guidance in WCAP-17096-NP, Revision 3, Appendix E, W-ID Item 9, for thermal shield flexures does not include the inspection guidance from Technical Bulletin (TB)-19-5 that Westinghouse issued after the determination that degraded thermal shield flexures were the root cause of the unexpected number of degraded BFBs at Salem, Unit 1 from the spring of 2019 outage (refer to the earlier discussion of the response to RAI-13 Part (c) in this SE). In the PWROG's response to RAI-13, Part (g), the PWROG explained that Revision 0 of TB-19-5 is referenced in MRP-227, Revision 1-A and that Revision 1 of TB-19-5 will be referenced in MRP-227, Revision 2 instead of in WCAP-17096-NP, Revision 3, Appendix E, W-ID Item 9, because the MRP-227 document in general (not the acceptance criteria methodology in the WCAP-17096 document in general) is associated with the inspection of thermal shield flexures and support block cap screws. The PWROG also stated that the thermal shield flexure methodology in WCAP-17096-NP, Revision 3, Appendix E, W-ID Item 9, postulates the existence of failed flexures and the likelihood that one failed flexure would result in a cascading failure of additional flexures. The NRC staff reviewed the response and WCAP-17096-NP, Revision 3, Appendix E, W-ID Item 9, and determined that the basis for not including the TB-19-5 inspection guidance in WCAP-17096-NP, Revision 3, Appendix E, W-ID Item 9, is acceptable because the NRC staff has confirmed that Item W9 in Table 4-3 of MRP-227, Revision 1-A, already references TB-19-5 as an inspection standard for W-design thermal shield flexures. The item analyzed in RAI-13 Part (g) is resolved.

Based on the discussion above, the NRC staff finds that the PWROG has adequately dispositioned relevant OE with respect to cracking in W-design BFBs and thermal shield flexures.

### 3.4.7 Acceptance Criteria Methodology and Data Requirements Used for the Evaluation of Component Void Swelling or Distortion

The NRC staff verified that Section 3.5 of the subject TR included the PWROG's bases for using visual inspection techniques as a method for detecting changes in dimension that may occur in PWR RVI components as a result of void swelling or distortion. Section 3.5 also included the bases for evaluating such effects, including the PWROG's discussions with respect to the detection and evaluation of swelling or distortion that may occur in the gap areas of CE-designed core shrouds that are fabricated in two vertical sections, as stated in WCAP-17096-NP, Revision 3, Appendix C, CE-ID Item 4a. The PWROG stated that the evaluation of the gap areas of the shrouds "will require additional sensitivity studies to relate the swelling level to the predicted distortion and gap opening in the structure," and that the sensitivity studies may be necessary "to evaluate the current condition,



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predict future effects, and determine aging management strategies, which are beyond the scope of the TR."

In addition, the NRC staff observed that the PWROG did not address any acceptance criteria or data analysis criteria for managing component changes in dimension due to void swelling or distortion in the subject TR. The NRC staff also observed that, as of the date of this SE, there has been little industry-reported OE with reported indications of changes in dimension, swelling, or distortion in W-design, CE-design, or B&W-design RVI components that the EPRI MRP has identified as being potentially susceptible to swelling or distortion effects. The NRC staff addressed this informational gap with the issuance of September 21, 2020, RAI-11 (Reference 23), in which the staff asked the PWROG to clarify whether the intent of Section 3.5 of the subject TR is for swelling-related distortion assessments to be the responsibility of licensees with the implementation of component analysis and acceptance criteria for these detected aging effects.

In the PWROG's response to RAI-11 (Reference 25), revisions were made to Section 3.5 of the subject TR to clarify that "a plant-specific plan should be developed for evaluating mitigating void swelling." Based on the PWROG's response to RAI-11 and its amendment to Section 3.5 of the subject TR, the NRC staff finds that the scope of the subject TR is limited in that it does not include any acceptance and data analysis criteria for evaluating component distortion, or void swelling, or for managing changes in dimension due to distortion, or void swelling in the PWR "Primary" or "Expansion" category components that are within the scope of the subject TR. Because the PWROG is placing the responsibility for assessing potential RVI component void swelling or distortion impacts on the license holder of the PWR unit, the use of the subject TR for void swelling or distortion evaluation objectives is subject to L-02 (WCAP-17096-NP, Revision 3), as established and defined in Table A-1 of Attachment A in this SE, "List of Limitations and Conditions" for WCAP-17096, Revision 3. Based on the addition of L-02 (WCAP-17096-NP, Revision 3), the item discussed in RAI-11 is resolved.

L-02 (WCAP-17096-NP, Revision 3):
The scope of WCAP-17096-NP, Revision 3, is limited in that it does not include any acceptance and data analysis criteria for evaluating component distortion or void swelling, or for managing changes in dimension due to distortion or void swelling in the PWR "Primary" or "Expansion" category components that are within the scope of WCAP-17096-NP, Revision 3. If distortion or changes in dimension due to void swelling is detected in an RVI component, a licensee using this TR will need to address distortion or changes in dimension due to void swelling on a component-specific basis per the acceptance criteria and corrective actions program element bases of its PWR Vessel Internals Programs and its 10 CFR Part 50, Appendix B program.

### 3.4.8 Combustion Engineering Design Components Assessed by Fatigue Evaluations

In Section 3.6 of the subject TR, the PWROG discussed how aging management of three CE-design "Primary" category components may be achieved through use of component fatigue or cyclical loading analyses. The NRC staff determined that, even though the changes to Section 3.6 of the subject TR represent "Administrative Edits" of the section, the process described in the subject TR warrants further evaluation and discussions with respect to a licensee's basis (for CE-designed plants only) for managing these CE-design components during an initial period of extended operation (i.e., licensed 40 to 60 years) or a subsequent period of extended operation (i.e., licensed 60 to 80 years).



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The three CE-design “Primary” category components that are subject to the EPRI MRP’s fatigue screening criteria in MRP-227, Revision 1-A, and to the PWROG’s corresponding statements in Section 3.6 of the subject TR and the CE-ID Items for the components in the subject TR Appendix C are:

- CSB flexure weld located in the CSB assembly of the plant design,
- core support plate in the lower support structure of the plant design, and
- fuel alignment plate in the upper internals assembly of the plant design

The NRC staff observed that the PWROG’s basis for evaluating these components in Section 3.6 of the subject TR (and in the referenced CE-ID item assessments in the subject TR Appendix C) are consistent with those established in the MRP-227, Revision 1-A. The EPRI MRP criteria establish the need for performing enhanced visual (EVT-1) examinations of the referenced CE-design “Primary” category component only if a fatigue analysis of the specified component in the CLB cannot demonstrate adequate screening against fatigue impacts. However, the concept behind the EPRI MRP’s criteria assumes that the CLB for referenced CE-design “Primary” category components will include at least one form of fatigue or cyclical loading analysis for the fatigue screening objective and that the analysis will be sufficient to demonstrate that: (1) initiation of a fatigue-induced macro crack or fatigue growth coalescence of two or more micro-cracks into a detectable macrocrack is unlikely to occur at the facility for the licensing period of interest, (2) the amount of analyzed and projected growth of a detected or postulated macro-crack in the component will remain within an specified allowable limit on stable flaw size through the end of the licensing period of interest, or (3) the fracture toughness values for the component will not be less than a specified lower-bound limit on acceptable fracture toughness values for the component.[10] However, the NRC staff also observed that there were specific regulatory and technical nuances on these “fatigue”-related “Primary” bases that require further explanations in this SE.

Specifically, the NRC staff observed that the EPRI MRP’s I&E bases defined in MRP-227, Revision 1-A, Table 4-2 (and the corresponding PWROG’s discussions in the subject TR in Section 3.6 and in Appendix C, CE-ID Items 7, 8, and 9), did not define the specific types of fatigue or cyclical loading analyses that could be used for performing the referenced “fatigue” screening objective for the specified component; however, the NRC staff acknowledges that analysis may be a cycle dependent cumulative factor analysis, fatigue waiver or exemption analysis, crack growth or flaw growth analysis, or fracture mechanics analysis. Due to the cyclical nature of these analyses, the NRC staff acknowledges that these types of analyses are often identified by LR/SLR applicants as being TLAAs for their LRAs or SLRAs, which is consistent with the statement in footnote 2 of the subject TR.

The NRC staff also observed that EPRI MRP’s criteria for these components are less clearly defined if the CLB for the specified component does not include at least one type of fatigue or cyclical loading analysis that could be credited for the referenced fatigue screening objective. Furthermore, to bypass the “Primary” EVT-1 examinations of CE-design CSB flexure welds, screening of the flexure welds for stress corrosion cracking (SCC) would also need to be performed, in addition to screening of fatigue, since the EPRI MRP identifies that CSB flexure welds are susceptible to both fatigue and SCC in Item C7 of MRP-227, Revision 1-A, Table 4-2. The NRC staff observed that the PWROG did not

[10] For example, a component cumulative usage factor (CUF) analysis or ASME Section III fatigue waiver or exemption analysis in the CLB could be used for satisfying the first numbered clause (1) in this sentence. Similarly, a component, cycle-dependent fatigue flaw growth analysis could be used to satisfy the second numbered clause (2) in this sentence, and a component, cycle-dependent linear elastic or elastic plastic fracture mechanics analysis could be used to satisfy the third numbered clause (3) in this sentence.



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address the susceptibility of fatigue and SCC on welds in Section 3.6 of the subject TR (even though the PWROG did account for it in WCAP-17096-NP, Revision 3, Appendix C, CE-ID Item 7, for the CSB flexure welds). The NRC staff addressed these informational gaps with the issuance of September 21, 2020, RAI-08, Parts (a), (b), and (c) (Reference 23). In the PWROG's final responses to RAI-08, Parts (a), (b), and (c) transmitted on January 14, 2022 (see Reference 25), revisions were made to the subject TR in Section 3.6 and in Appendix C, CE-ID Items 7, 8, and 9, to include fatigue screening evaluation criteria for CE-design core support barrel (CSB) flexure welds, core support plates, and fuel alignment plates and SCC screening evaluation criteria for CE-design CSB flexure welds.

Based on the changes to the responses to RAI-08 Parts (a), (b), and (c) (Reference 25), from the previous versions, the NRC staff observed that the PWROG is now placing fatigue screening needs for the referenced CSB flexure welds, core support plates, and fuel alignment plates, and the SCC screening needs for CSB flexure welds outside the scope of the subject TR instead of amending the report to include the appropriate acceptance criteria methodology and data requirements. Specifically, for the Item 7, 8, and 9, assessments evaluated in the PWROG's response to RAI-08 (Reference 25), the PWROG removed the component-specific screening analysis assessment criteria for evaluating fatigue and SCC in CE-design core barrel flexure welds and fatigue in CE-design core support plates and upper fuel alignment plates from the scope of the referenced ID item assessments in Appendix C of the subject TR.

In order to address how to evaluate component screening assessment items that are outside of the scope of the subject TR's methodology, the NRC staff has identified a limitation on the RAI-08 fatigue and SCC screening topics and the lack of defined fatigue screening bases in the subject TR when referencing CSB flexure welds, core support plate and fuel alignment plates components (as assessed in CE-ID Items 7, 8, and 9) or defining SCC screening bases for the CSB flexures welds (CE-ID Item 7) in the subject TR. Therefore, these screening topics are the subject of L-03 (WCAP-17096-NP, Revision 3) and the clarifications made in the limitation, as established are defined in Table A-1 of Attachment A of this SE. Based on the addition of this limitation, the items discussed in RAI-08, Parts (a), (b), and (c), are resolved.

L-03 (WCAP-17096-NP, Revision 3):
The scope of the WCAP-17096-NP, Revision 3, is limited in that it no longer includes or defines the acceptance criteria or data analysis criteria for performing the fatigue and SCC screening analyses of CE-design core barrel flexure welds or fatigue screening analyses of CE-design core support plates and upper fuel alignment plates. Consistent with the guidelines in MRP-227, Revision 1-A, a licensee using the subject TR should address the acceptance criteria and data analysis criteria for performing the component-specific screening analyses of CE-design core barrel flexure welds, core support plates, and upper fuel alignment plates on a unit-specific or site-specific case-by-case basis.

### 3.4.9 Resolution for "Technical Justification for Removal" in Topical Report WCAP-17906, Revision 3

In Appendix A, of the subject TR, the PWROG identified that a "*technical justification for removal*" may be necessary for specific types of B&W designed components. The PWROG's statements apply to the component types in the following B&W-ID Item assessments in Appendix A:

- Locking Devices for Baffle to Former Bolts and Internal Baffle to Baffle Bolts



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- Alloy X-750 Dowel-to-Guide Blocks Welds
- Upper Grid Fuel Assembly Dowel-to-Support Pad Welds
- Lower Grid Fuel Support Pads Items
- Alloy X-750 Dowel-to-Support Pad Welds

In RAI-09, (Reference 23), the NRC staff informed the PWROG that it was not evident whether the referenced statement was addressing a physical removal of the referenced component through licensee performance of a justifiable design modification or removal of the component from either the "Primary" or "Expansion" category inspection category in MRP-227, Revision 1-A and recategorization as "No Addition Measures" components for the program. In the PWROG's response to RAI-09 (Reference 26), the PWROG clarified that the referenced statement is made in reference to a physical removal of the component in reference. Given that physical removal of the referenced components from the plant design would require the licensee to implement the design change through a 10 CFR 50.59 review, the NRC staff finds the explanation acceptable and RAI-09 is resolved.

3.4.10 Fracture Toughness Considerations

In Appendices C and E of the subject TR, the PWROG referenced the following four neutron fluence-dependent, lower bound, linear elastic fracture mechanics (LB-LEFM) acceptance criterion values for specified SS base metal or weld components:

- 150 ksi-√inch for components with project neutron fluence exposures ≤ $3x10^{20}$ n/cm$^2$ (E > 1.0 MeV; ≤ 0.5 dpa) at the end of the service period of interest[11]

- 112 ksi-√inch for components with project neutron fluence exposures > $3x10^{20}$ n/cm$^2$ and ≤ $3x10^{21}$ n/cm$^2$ (E > 1.0 MeV; > 0.5 dpa and ≤ 5 dpa) (E > 1.0 MeV; ≤ 0.5 dpa) at the end of the service period of interest

- 50 ksi-√inch for components with project neutron fluence exposures > $3.0x10^{21}$ n/cm$^2$ and ≤ $1.0x10^{22}$ n/cm$^2$ (E > 1.0 MeV; > 5 dpa and ≤ 15 dpa) at the end of the service period of interest

- 34.6 ksi-√inch for components with project neutron fluence exposures > $1.0x10^{22}$ n/cm$^2$ (E > 1.0 MeV; > 15 dpa) at the end of the service period of interest

For the referenced 150 ksi-√inch, 112 ksi-√inch, and 50 ksi-√inch LB-LEFM values, the NRC staff observed that the applicant was indirectly referencing the EPRI BWRVIP-100, Revision 1-A (Reference 41)[12] as the source document basis for the referenced LB-LEFM values. However, the NRC staff also observed that the PWROG did not provide the source document for the referenced LB-LEFM value of 34.6 ksi-√inch because BWRVIP-100, Revision 1-A does not cover that lower bound LEFM fracture toughness value. In RAI-10 (Reference 23), the NRC staff asked for further justification for use of the 34.6 ksi-√inch lower bound LEFM fracture toughness value for SS base metal or weld components. In the PWROG's response to RAI-10 (Reference 25), it cited EPRI MRP Report MRP-

[11] For example, the service period of interest could be at the end of the first renewed period of extended operation if made in reference to a first license renewal extended period of operation, or the end of the augmented inservice inspection interval of interest if being used for justification of a component reinspection interval that differs from the 10-year reinspection interval bases in MRP-227, Revision 1-A.

[12] Updated bases for this EPRI report were approved in EPRI Proprietary Report No. 3002008388 (Reference 37).



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211[13] as the basis for the 34.6 ksi-√inch lower bound LEFM fracture toughness value. In addition, the PWROG amended the source reference for the 34.6 ksi-√inch line item entry in the fracture toughness table for CE-ID Item 2, Core Shroud Assembly (Welded), Core Shroud Plate-Former Plate Weld as MRP-211 and indicated that there are plans to amend the corresponding 34.6 ksi-√inch based line item entries of the fracture toughness tables in the corresponding ID item assessments to include the MRP-211 in the following:

- CE Core Shroud Assembly (Welded) - Core Shroud Plate-to-Former Plate Weld
- CE Core Shroud Assembly (Welded) - Remaining Axial Welds (CE-design units with welded core shrouds assembled in two vertical sections)
- CE Core Shroud Assembly (Welded) - Shroud Plates
- CE Core Shroud Assembly (Welded), Remaining Axial Welds (CE-design units with welded core shrouds assembled with full height shroud plates)
- CE CSB Assembly, UGW
- CE CSB Assembly, UAW
- CE CSB Assembly, MGW
- CE CSB Assembly, MAW
- CE CSB Assembly, LAW
- CE CSB Assembly, UFW
- CE CSB Assembly, LGW
- CE CSB Assembly, CSB Flexure Weld
- CE Lower Support Structure, Deep Beams (CE-design units with welded core shrouds assembled with full height shroud plates)
- CE Lower Support Structure, Lower Core Support Beams (All CE-design units except those with welded core shrouds assembled with full height shroud plates)
- W Core Barrel Assembly, UFW
- W Core Barrel Assembly, UGW
- W Core Barrel Assembly, UAW
- W Core Barrel Assembly, LGW
- W Core Barrel Assembly, MAW
- W Core Barrel Assembly, LAW
- W Core Barrel Assembly, LFW

Since the PWROG has yet to adjust the lower bound fracture toughness entries in the ID items in the subject TR, the staff has addressed the gap as part of Condition C-03. From a technical perspective, the NRC staff observed that the 34.6 ksi-√inch line item entries in the fracture toughness tables of the applicable ID item assessments continue to identify that the 34.6 ksi-√inch based value becomes applicable at a projected neutron fluence range of *"> 1x10$^{22}$ n/cm$^2$ (E > 1.0 MeV),"* whereas as the MRP-211 document actually cites that the 34.6 ksi-√inch lower bound LEFM value is applicable at a neutron fluence range somewhat lower than that (i.e., MRP-211 reports it as being applicable at projected neutron fluence exposures in excess of 6.7x10$^{21}$ n/cm$^2$ (E > 1.0 MeV) or projected doses in excess of 10 dpa). The NRC staff also observed an administrative error in the revised response to RAI-10 for the associated amendments of the fracture toughness tables in the February 8, 2022, RAI response, where the PWROG cited the applicable MRP report in the revised/updated footnote 2 of the tables as being "*MRP-221,*" and not as "*MRP-211*" as indicated in the response to RAI-10.

[13] For the objectives of the staff's SE, the staff is referencing Revision 1 of the MRP-211 Report (Reference 35).



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Additionally, the NRC staff observed that, in the fracture toughness tables of the applicable ID items of , Appendices C and E of the subject TR, the PWROG uses EPRI proprietary version BWRVIP-100, Revision 1-A as the source record for the cited lower bound LEFM value of 150 ksi-√inch, which is being applied for SS components with projected neutron fluence exposures < $3x10^{20}$ n/$cm^2$ (E > 1.0 MeV). Similarly, in the fracture toughness tables of the applicable ID items in Appendices C and E of the subject TR, the PWROG uses BWRVIP-100, Revision 1-A as the source record for the lower bound LEFM facture toughness values of 112 ksi-√inch (applicable to SS components with projected component fluence exposures > $3x10^{20}$ n/$cm^2$ and ≤ $3x10^{21}$ n/$cm^2$ [E > 1.0 MeV]) and 50 ksi-√inch (applicable to SS components with projected component fluence exposures > $3x10^{21}$ n/$cm^2$ and ≤ $1x10^{22}$ n/$cm^2$ [E > 1.0 MeV]). The NRC staff observed that in EPRI BWRVIP Letter No. 2021-030 (Reference 42), the EPRI BWRVIP reported new fracture toughness data may create some potential issues with the lower bound LEFM fracture toughness values reported for SS components with projected neutron fluence exposures in excess of $5x10^{20}$ n/$cm^2$ (E > 1.0 MeV).

Based on the discussion in this section, NRC staff concluded that there may be some uncertainty with the accuracy of the fluence ranges reported for lower bound LEFM fracture toughness values reported in the fracture toughness tables provided in Appendices C and E of the subject TR. Therefore, the licensee should take actions to confirm that all reported lower bound LEFM fracture toughness values and the fluence ranges reported for the lower bound LEFM fracture toughness values in the subject TR are valid and accurate or else amend them accordingly. The selection of the applicable lower bound fracture toughness value(s) should include an appropriate justification and identified source record and should be reconciled with any updated industry reports or records used to correlate lower bound fracture toughness value(s) and the most recent neutron fluence range(s) reported for the specified fracture toughness value(s).

The NRC staff has identified this item as Condition C-03 (WCAP-17096-NP, Revision 3), as reflected in Table A-1 of Attachment A.

C-03 (WCAP-17096-NP, Revision 3):
Until such time that the EPRI or PWROG can resolve the EPRI BWRVIP Letter No. 2021-020 Part 21 issue on neutron fluence ranges associated with specified lower bound fracture toughness values reported in an updated version of MRP-210 or MRP-211 (or alternate industry record), licensees will need to justify any lower bound fracture toughness values used as inputs in component-specific flaw tolerance analyses. The selection of the applicable lower bound fracture toughness value(s) should be reported and justified in a site-specific record and reconciled with any updated industry reports or records used to correlate lower bound fracture toughness value(s) and the most recent neutron fluence range(s) reported for the specified fracture toughness value(s).

### 3.4.11 Babcock and Wilcox-Design Components Currently Lacking a Formal Inspection Standard or Aging Effect Analysis Methodology

For B&W-design RVI components corresponding to the applicable ID item in Appendix A of the subject TR, the NRC staff observed that the ID item assessments indicated that the components will either receive “Primary” VT-3 visual inspections or potential “Expansion” based VT-3 visual inspections if the “expanded” inspections of the Expansion category component type is found to be applicable from results of visual inspections performed in the linked “Primary” component. The NRC staff observed that the PWROG included the following statement from Appendix A Section 1.2



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(Control Rod Guide Tube Spacer Castings) in the subject TR for a number of B&W component ID item assessments:

> *"The NDE inspection standard could be developed generically."*

The NRC staff observed that for components in the applicable sections of the Appendix A of the subject TR, that included statements of this nature, the report did not firmly establish what the acceptance criteria methodology and data requirements would be for evaluating the results of VT-3 visual inspections applied to the components (or that might be applied to components that are designated as Expansion category components for the program). Therefore, the NRC staff addressed this information gap in RAI-12 (Reference 23).

In the PWROG's response to RAI-12 (Reference 26), it stated that the referenced B&W-ID Items include general provisions for identification of: (1) "*what visual examination wear indications are considered rejectable and what would require additional examination and evaluation*," (2) "*any additional examination results that are anticipated*" and the "*general acceptance criteria for the additional items expected to be in the VT-3 visual examination field of vision.*" However, the PWROG admitted that these statements did not fully define the acceptance and data analysis criteria for the components in these B&W-ID item assessments. Therefore, the PWROG proposed to replace the statements with the following additional action for B&W-ID Items:

> *"The VT-3 exam acceptance criteria are "no relevant conditions." Relevant conditions for the B&W Items of interest are defined in Section 5 of MRP-227-1A. These relevant conditions constitute discontinuities or imperfections (e.g., loose or missing parts, distortion, corrosion, wear, etc.), which are readily detectable with the VT-3 technique. Note that the VT-3 exam method meets the requirements of ASME B&PV Code Section XI, as specified and conditioned by 10 CFR 50.55a, and the MRP-228 Inspection Standard."*

For B&W components in those referenced B&W-ID item assessments calling for VT-3 visual inspections (e.g., the CRGT spacer castings in B&W-ID Item A.1.2, etc.), the PWROG's revised data analysis and acceptance criteria basis in the above reference statement creates a limitation on the scope of the report because it defines "*relevant VT-3 visual detected conditions*" as being those that are defined in MRP-227, Revision 1-A and not on those that would be included and defined in the applicable B&W-ID items referenced above. Furthermore, Section 5 of the MRP-227, Revision 1-A only gives examples "*relevant VT-3 visual detected conditions*" for aging mechanism parameters inspected by VT-3 visual methods and in many cases leaves the full criteria for defining "*relevant VT-3 visual detected conditions*" up to the licensees implementing their PWR Vessel Internals Programs (or equivalent named program for the internals in the CLB). Therefore, the PWROG is holding the licensee responsible for adding the inspected component in the NRC licensee's corrective action program based on: (1) a licensee's finding of a relevant condition, (2) how the licensee defines relevant conditions for the applicable component to match up to the general "*relevant condition"* bases for the component in Table 5-1 of the MRP-227, Revision 1-A (e.g. as generally defined Line Item C2 for the CRGT spacer castings in Table 5-1 of the MRP-227, Revision 1-A), and (3) how many relevant conditions would need to be detected in order to expand the VT-3 visual inspection basis to any Expansion category components that may be linked to the referenced "Primary" component.

Given that the PWROG is no longer in the position of defining the relevant VT-3 conditions or defining the acceptance criteria methodology and data requirements for relevant VT-3 conditions that may potentially be detected in the referenced B&W-design components of the applicable B&W-ID items, the subject TR is limited in that it places the responsibility on the licensee holding the first renewed or



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subsequent renewed operating licensee of the B&W designed PWR to define the relevant VT-3 conditions and the acceptance criteria for relevant conditions of the referenced B&W components. In accordance with its operating license, the licensee should define these aspects of the programs as part of its PWR vessel internals AMP. This is identified as L-04, which is explained in Table A-1 of Attachment A of this SE. Based on the addition of this Limitation, RAI-12 is resolved.

L-04 (WCAP-17096-NP, Revision 3):
The scope of the WCAP-17096-NP, Revision 3, is limited in that it does not define any acceptance criteria or data analysis criteria for evaluating relevant conditions in B&W-design RVI components that are inspected using VT-3 visual inspection methods. Consistent with the guidelines in MRP-227, Revision 1-A, the acceptance criteria and data analysis criteria for evaluating relevant conditions in B&W-design "Primary" or "Expansion" category components that are subject to VT-3 inspections are to be addressed on a component-specific case-by-case basis by a licensee using WCAP-17096-NP, Revision 3.

### 3.4.12 Westinghouse Design Control Rod Guide Tube Support Plates

The PWROG established its updated set of acceptance criteria methodology and data requirements for W-design CRGT guide plates (guide cards) in WCAP-17096-NP, Revision 3 Appendix E, W-ID Item 1. The NRC staff found that the acceptance criteria methodology and data requirements for CRGT guide cards in W-ID Item 1 to be acceptable for implementation because they were consistent with those established for the CRGT guide cards in Revision 2 of the TR, with one exception that needed further explanations from the PWROG. Specifically, the NRC staff observed that, in W-ID Item 1 assessment criteria bases, the PWROG included the two following data analysis criteria for the CRGT guide card evaluations:

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

The NRC staff observed that the data analysis criteria did not include the following additional (third) data analysis criterion for W-design CRGT guide cards that was included in the corresponding "Data Requirements" subsection of the applicable ID item:

> *"Guide tube operational effective full power years (EFPYs) at the time of the inspection or measurement."*

The NRC staff addressed this information gap with the issuance of RAI-14 (see Reference 23) which requested that the PWROG reconcile the differences in the data analysis criteria given on the "Data Requirements" section of W-ID Item 1 in Appendix E of the subject TR, from the corresponding data analysis criteria for the CRGT guide card components in Appendix E of WCAP-17096-NP, Revision 2.

In the PWROG's response to RAI-14 (see Reference 25), it stated that the inclusion of W-ID Item 1 data analysis criterion on "innermost hole ligament wear" was inadvertently duplicated in the "Data Requirements" section of the ID item, and that the "Data Requirements" section should have included the bulleted item on "guide tube operational effect full power years." The PWROG amended W-ID



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Item 1 in the subject TR to include the applicable statement from WCAP-17096-NP, Revision 3, cited above. Based on this amendment of the subject TR, the NRC staff finds the acceptance criteria methodology and data requirements in W-ID Item 1 of WCAP-17096-NP, Revision 3, Appendix E, to be acceptable and RAI-14 is resolved.

### 3.4.13 Relevant Operating Experience and Industry Interim Guidance Methodology Not Addressed in Topical Report WCAP-17096, Revision 3

The NRC staff observed that based on EPRI Letter No. MRP 2019-023 dated September 3, 2019 (Reference 40), EPRI established Interim Guidance for the inspections of middle axial welds (MAWs) and lower axial welds (LAWs) located in the core barrels of W-design PWRs or core support barrels of CE-design PWRs. The MRP 2019-023 letter included and docketed EPRI's formal interim guidelines for inspecting the MAWs and LAWs in EPRI Letter No. MRP 2019-009 (Reference 40). Attachment 1 to MRP 2019-009 provided EPRI's rationale for creating "Needed" interim inspection guidelines for these MAWs and LAWs.

The NRC staff observed that the interim guidelines in EPRI Letter 2019-009 did not formally include or establish any applicable acceptance criteria for the one-time VT-3 visual examinations that would be applied to the MAWs and LAWs using EPRI's interim guidance criteria for the components. The NRC staff also observed that the information in WCAP-17096-NP, Revision 3, Section 2.3, and in Appendix C, CE-ID Items 6.1 and 6.2, and WCAP-17096-NP, Revision 3, Appendix E, W-ID Items 4.2 and 4.3, do not reference the existence of the EPRI MRP 2019-009 and MRP 2019-023 letters or provide any acceptance criteria for the one-time VT-3 visual examination that is recommended for the MAWs and LAWs in MRP 2019-009. The NRC staff addressed these informational gaps with the issuance of RAI-06, Parts (a) and (b) (Reference 23).

In the PWROG's complete response to RAI-06, Parts (a) and (b) (see Reference 25), it clarified that the interim guidelines in MRP-2019-009 that require a one-time VT-3 of the core support barrel/core barrel MAWs and LAWs are only cited as NEI-03-08 "Good Practice" protocols and are not issued as NEI 03-08-established "Needed" or "Mandatory" criterion under the umbrella of the "Needed" criterion of the I&E tables in Sections 4 and 5 of the MRP-227, Revision 1-A. Based on these clarifications, the NRC staff finds that the PWROG has provided a sufficient basis regarding why the subject TR does not include any acceptance criteria methodology and data requirements in WCAP-17096-NP, Revision 3, Appendix C, CE-ID Items 6.1 and 6.2, or in Appendix E, W-ID Items 4.2 and 4.3, for the one-time VT-3 examinations that may be applied as "*Good Practices*" for core support barrel/core barrel MAWs and LAWs. The items discussed in RAI-06, Parts (a) and (b) are resolved.

## 4.0 REFERENCING TOPICAL REPORT WCAP-17096, REVISION 3 IN THE CURRENT LICENSING BASIS

WCAP-17096-NP, Revision 3, and its relationship to the CLB is discussed in Section 3.2 of this SE. In addition, NRC staff has identified Limitation L-01 (WCAP-17096-NP, Revision 3) as discussed in this SE and presented in detail in Attachment A. NRC licensees applying the subject TR to their licensing basis or design basis should follow Limitation L-01 (WCAP-17096-NP, Revision 3) in alignment with their specific type of operating license that is currently in effect for the CLB. Section 3.2.1 of this SE provides information for holders of original operating licenses. Similarly, Section 3.2.2 of this SE gives details for holders of first renewed operating licenses, and Section 3.2.3 of this SE explains the process for holders of subsequent renewed operating licenses.



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## 5.0 LIMITATIONS AND CONDITIONS

As a result of its review of the subject TR and supplemental information, the NRC staff has identified Limitations and Conditions in Section 3.0 of this SE which are associated with the implementation of this TR. Evaluations for certain components will require plant specific analyses or generic analyses that will need to be submitted to the NRC staff in the event that the inspection results do not meet the acceptance criteria or trigger the expansion criteria methodology in MRP-227. As stated in Section 3.0, limitations and conditions identify any additional plant-specific action items that will be needed to support the NRC's staff's review of a request to implement the TR. Limitations and conditions identify the items that will need to be addressed in subsequent individual plant applications which plan to reference the subject TR. Based on its review of the subject TR and supplemental information, the NRC staff identified four Limitations and three Conditions in Section 3.0 of this SE regarding the implementation of the TR.

A complete detailed overview of all limitations and conditions are available in Attachment A to this SE, Table A-1, "Summary of Limitations and Conditions," for Topical Report WCAP-17096, Revision 3.

## 6.0 SUMMARY AND CONCLUSIONS

The NRC staff has reviewed WCAP-17096-NP, Revision 3, and concludes that the report, as modified by the limitations and conditions in Table A-1 of Attachment A, describes acceptable methods for assessing aging degradation that may be found in RVI components. For NRC licensees conducting inspections in accordance with MRP-227, Revision 1-A, and using the appropriate evaluation procedures, the NRC staff concludes that WCAP-17096-NP, Revision 3, is acceptable for referencing in plant-specific RVI AMPs, as subject to the limitations and conditions in this SE.

## 7.0 REFERENCES

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2. Pressurized Water Reactor Owners Group (PWROG) letter to U. S. Nuclear Regulatory Commission, "Transmittal of the Response to Request for Additional Information RAIs 1-12 and 14 Associated with WCAP-17096, Revision 3, "Reactor Internals Acceptance Criteria Methodology and Data Requirments" (PA-MSC-1567)," April 8, 2021 (ADAMS Package Accession No. ML22187A141).

3. Powell, Michael, PWROG letter to U.S. Nuclear Regulatory Commission, "Transmittal of the Response to Request for Additional Information RAIs Associated with WCAP-17096, Revision 3, "Reactor Internals Acceptance Criteria Methodology and Data Requirements, (PA-MSC-1567)," January 14, 2022 (ADAMS Package Accession No. ML22187A149).

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11. Electric Power Research Institute (EPRI) to Materials Reliability Program, "Materials Reliability Program: Pressurized Water Reactor Internals Inspection and Evaluation Guidelines (MRP-227-A) EPRI, Palo Alto, CA 2011; 1022863 Transmittal of Interim Guidance," February 18, 2014 (ADAMS Accession No. ML14274A372).

12. U. S. Nuclear Regulatory Commission (NRC) letter to EPRI, "Final Safety Evaluation of WCAP-17096-NP, Revision 2, Reactor Internals Acceptance Criteria Methodology and Data Requirements," May 3, 2016 (ADAMS Accession No. ML16061A243).

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

14. U. S. Nuclear Regulatory Commission (NRC) letter to Electric Power Research Institute (EPRI), "U. S. Nuclear Regulatory Commission Approval Letter for the Electric Power Research Institute Topical Report for WCAP-17096-NP-A, Revision 2, Reactor Internals Acceptance Criteria Methodology and Data Requirements," January 3, 2017 (ADAMS Accession No. ML16271A001).

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29. U. S. Nuclear Regulatory Commission (NRC), "Regulatory Guide (RG) 1.147, Revision 20 - Inservice Inspection Code Case Acceptability ASME Section XI, Div. 1," December 17, 2021 (ADAMS Accession No. ML21181A222).

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31. Electric Power Research Institute (EPRI) Letter No. MRP 2016-021, "Transmittal of NEI 03-08 "Needed" Interim Guidance Regarding Baffle Former Bolt Inspections for Tier 1 Plants as Defined in Westinghouse NSAL 16-01," July 27, 2016 (ADAMS Accession No. ML16210A006).

32. Electrical Power Research Institute (EPRI) Letter No. MRP 2017-009, "Transmittal of NEI 03-08 "Needed" Interim Guidance Regarding Baffle Former Bolt Inspections for Tier 1 Plants as Defined in Westinghouse NSAL 16-01, Rev. 1," March 15, 2017 (ADAMS Accession No. ML17087A106).

33. PWR Owners Group Letter OG-18-226 “Transmittal of PWROG-18034-P and PWROG-18034-NP Revision 0, ‘Updates to the Methodology in WCAP-15029-P-A, Revision 1, 'Westinghouse Methodology for Evaluating the Acceptability of Baffle-Former-Barrel Bolting Distributions Under Faulted Load Conditions,' PA-MSC-1519," October 31, 2018 (ADAMS Package Accession No. ML18306A583). REFERENCE NOTE: Errata to the WCAP-15029-P-A, Revision 1 and WCAP-15030-NP-A, Revision 0 are given in ADAMS Package Accession No. ML18306A789.

34. PWR Owners Group letter to U.S. Nuclear Regulatory Commission, "Submittal of PWROG-17071-NP, Revision 0, ‘WCAP-17096-NP-A Interim Guidance’ to the NRC for Information Only (PA-MSC-1567)," July 12, 2018 (ADAMS Accession Nos. ML18204A165 and ML18204A166).

35. Electric Power Research Institute (EPRI) Report No. 3002010270, "Materials Reliability Program: PWR Internals Age-Related Material Properties, Degradation Mechanisms, Models and Basis Data − State of Knowledge (MRP-211, Revision 1)," October 2017 (ADAMS Package Accession Nos. ML17361A168/ML17361A191 - Proprietary/Non-Public); and ML17361A189, - Non-Proprietary/Public).

36. NRC Agency Record providing “Office of Nuclear Reactor Regulation Staff Assessment of Electric Power Research Institute NEI 03-08, Revision 2, ‘NEEDED Interim Guidance Regarding Baffle-Former Bolt Inspections in Westinghouse Design Pressurized Water Reactors,’” November 20, 2017 (ADAMS Accession No. ML17310A861).

37. Westinghouse Electric Company Proprietary Class 2, "Westinghouse Baffle-Former Bolt Predictive Methodology," September 18, 2019 (ADAMS Accession Nos. ML19242B369 and ML19274F482 - Proprietary/Non-Public; ML19274F472 -Non-Proprietary/Public).

38. U. S. Nuclear Regulatory Commission, letter to Westinghouse Owners Group Steering Committee, "Safety Evaluation of Topical Report WCAP-15029, Westinghouse Methodology



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for Evaluating the Acceptability of Baffle-Former-Barrel /bolting Distributions Under Faulted Load Conditions," November 10, 1998 (ADAMS Accession No. ML20195C636).

39. Westinghouse Electric Company, LLC, Attachment 1 to January 14, 2022, letter to PWR Owners Group, "PWR Owners Group Responses to NRC Request for Additional Information (RAIs) on WCAP-17096-NP., Rev. 3," RAI 13, January 13, 2022 (ADAMS Accession No. ML22039A146).

40. Electric Power Research Institute (EPRI), "Transmittal of MRP-227-A Related Interim Inspection Guidance Regarding PWR Core Barrel," MRP-2019-023 (BWRVIP 2019-009), September 3, 2019 (ADAMS Accession No. ML19249B102).

41. Electric Power Research Institute (EPRI), "2017 Technical Report - BWRVIP-100, Revision 1-A: BWR Vessel and Internals Project, Updated Assessment of the Fracture Toughness of Irradiated Stainless Steel for BWR Core Shrouds," February 2017 (ADAMS Accession No. ML17076A232 - Proprietary/Non-Public).

42. Electric Power Research Institute (EPRI), "BWRVIP Letter No. 2021-030 - Potential Non-Conservatism in EPRI Report, BWRVIP-100, Revision 1-A, 30020008388 and Impacted BWRVIP Reports," March 22, 2021 (ADAMS Accession No. ML21084A164).

43. Electric Power Research Institute (EPRI), "Transmittal of MRP-227-A-Related Interim Inspection Guidance Regarding PWR Core Barrel," September 3, 2019 (ADAMS Accession No. ML19249B102).

44. Pressurized Water Reactor Owners Group (PWROG), "Transmittal of the Response to Request for Additional Information (RAIs) Associated with WCAP-17096, Revision 3, Reactor Internals Acceptance Criteria Methodology and Data Requirements (PA-MSC-1567)," January 14, 2022 (ADAMS Accession No. ML22018A068).

45. Entergy Nuclear Operations, Inc., Palisades Nuclear Plant, "Unplanned Shutdown Due to Elevated Primary Coolant System Leak Rate and Permanent Plant Shutdown," May 23, 2022 (ADAMS Accession No. ML22143A937).

46. Electric Power Research Institute (EPRI) letter to U. S. Nuclear Regulatory Commission (NRC), "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," July 13, 2017 (ADAMS Accession No. ML17261B149).

47. Electric Power Research Institute (EPRI) Letter No. BWRVIP 2019-016, “White Paper on Suggested Content for PFM Submittals to the NRC,” February 27, 2019 (ADAMS Accession No. ML19241A545).”



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Attachments: A. Summary of Limitations and Conditions
B. Comment Resolution Table

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Date: July 24, 2023



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| Overview of PWROG RAI Response Transmittals | | | |
|---|---|---|---|
| **Document Title / Description** | **Document Date** | **Agency Record** | **Reference No.** |
| PWR Owners Group Transmittal of the Response to Request for Additional Information, RAIs Associated with WCAP-17096, Revision 3, "Reactor Internals Acceptance Criteria Methodology and Data Requirements" (PA-MSC-1567) | January 14, 2022 | ML22018A068 | Reference 44 |
| PWR Owners Group Transmittal Letter in Response of Request for Additional Information (RAIs) on WCAP-17096-NP, Revision 3 | January 14, 2022 | ML22039A143 | Reference 24 |
| PWR Owners Group – Response to NRC Requests for Additional Information (RAIs) on WCAP-17096-NP, Revision 3: (RAI-01; RAI-04; RAI-05; RAI-06; RAI-07 RAI-08; RAI-10; RAI-11 and RAI-14) | January 13, 2022 | ML22039A144 | Reference 25 |
| Westinghouse Response to RAI-13 on WCAP-17096-NP, Revision 3 | January 13, 2022 | ML22039A146 | Reference 39 |
| Framatome Responses to NRC WCAP-17096, Revision 3 (RAIs) Technical Report (RAI-02; RAI-03; RAI-09; RAI-12; and RAI-13 (b) | January 13, 2022 | ML22039A147 | Reference 26 |

**Please Note: On February 8, 2022, the PWROG resubmitted the January 13, 2022, RAI responses to NRC and the transmittal was added to the ADAMS docket. The February 8, 2022, transmittal superseded the January 14, 2022, transmittal, however the RAI document date of January 13, 2022, remains unchanged.*



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# Safety Evaluation, Attachment A

## Table A-1, Summary of Limitations and Conditions



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| NRC SAFETY EVALUATION FOR TOPICAL REPORT WCAP-17096-NP, REVISION 3, "Reactor Internals Acceptance Criteria Methodology and Data Requirements"<br>TABLE A-1: SUMMARY OF LIMITATIONS AND CONDITIONS | | | | | |
|---|---|---|---|---|---|
| Numerical Identifier for Limitations and Conditions | | | Limitations and Conditions | Rev. 3 Topical Report (TR) & Safety Eval. (SE) Sections | |
| Number | Topic / RAI | Technical Basis | | TR | SE |
| Limitation L-01(a)(b)(c) | Applicability of Revision 3 of the TR to the current licensing basis CLB | The generic applicability of Rev. 3 of the TR to the licensing basis for a given PWR unit is dependent on the following factors in the CLB for the unit:<br><br>(1) the type of plant operating license (i.e., original, first renewed, or subsequent renewed license) held by the NRC-licensed facility which owns the specified PWR unit,<br><br>(2) the type of aging management program (i.e., plant-specific AMP versus generic MRP-227-based AMP) being applied to the RVI components in the unit,<br><br>(3) the type of generic EPRI MRP-based program being applied as the basis for the AMP in the CLB, and<br><br>(4) the version of MRP-227 being applied to the program. | **Limitation 1 (L-01)(a)**:<br><br>*SE Section 3.2.1: Licensing Considerations- Comanche Peak, Diablo Canyon, and Watts Bar*<br>For PWR plants with an original operating license in a pre-submittal phase for first-time license renewal or currently undergoing an NRC review of a first-time license renewal application, NRC licensees may use the subject TR if they are implementing AMPs in accordance with MRP-227, Revision 1-A.<br><br>**L-01(b)**:<br><br>*SE Section 3.2.2: Holders of First Renewed Operating Licenses*<br>For PWR plants with first renewed operating licenses, NRC licensees may use the subject TR if they are implementing AMPs in accordance with MRP-227, Revision 1 -A.<br><br>**L-01(c)**:<br><br>*SE Section 3.2.3: Holders of Subsequent Renewed Operating Licenses*<br>For PWR plants with subsequent-renewed operating licenses, NRC licensees may use the subject TR only for those RVI components whose I&E criteria for the components are still consistent with MRP-227, Revision 1-A, and have not been modified by the results of an 80-year RVI gap analysis or a version of MRP-227 covering an 80-year aging assessment period. | Revision 3 Chapter 1 | 3.2, 3.2.1, 3.2.2, & 3.2.3 |



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| Number | Topic / RAI | Technical Basis | | TR | SE |
| Limitation L-02 | In Revision 3 of the TR, the PWROG's basis for evaluating VT-3 visual inspection results that provide evidence of void swelling or distortion in components. / RAI-11 | In the PWROG's response to RAI-11 (Reference 6 of this Final SE), which places the responsibility for evaluating component void swelling or distortion on the NRC licensee implementing its PWR Vessel Internals Program (or an equivalently named RVI AMP for the CLB). | **Limitation 2 (L-02)**:<br><br>The scope of WCAP-17096-NP, Revision 3, is limited in that it does not include any acceptance and data analysis criteria for evaluating component distortion or void swelling, or for managing changes in dimension due to distortion or void swelling in the PWR "Primary" or "Expansion" category components that are within the scope of WCAP-17096-NP, Revision 3. If distortion or changes in dimension due to void swelling is detected in an RVI component, a licensee using this TR will need to address distortion or changes in dimension due to void swelling on a component-specific basis per the acceptance criteria and corrective actions program element bases of its PWR Vessel Internals Programs and its 10 CFR Part 50, Appendix B program. | Revision 3 Section 3.5 | 3.4.7 |



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| Numerical Identifier for Limitations and Conditions | | | Limitations and Conditions | Rev. 3 Topical Report (TR) & Safety Eval. (SE) Sections | |
| Number | Topic / RAI | Technical Basis | | TR | SE |
| Limitation L-03 | Fatigue screening requests for CE-design core support plates in lower support structures and fueling alignment plates in upper internals assembly, and fatigue and stress corrosion cracking (SCC) screening requests for CE-design core support barrel (CSB) flexure welds.<br>/<br>RAI-08, Parts (a), (b), and (c) | In the PWROG's response to RAI-08 (Reference 24, this SE), NRC staff observed that the PWROG's fatigue screening requests for CSB flexure welds is outside the scope of the WCAP-17096-NP, Revision 3. Therefore, NRC staff has identified a limitation on the RAI-08 fatigue and SCC screening topics when licensees reference CSB flexure welds, core support plate and fuel alignment plates components in WCAP-17096-NP, Revision 3 (as assessed in CE-ID Items 7, 8, and 9) or defining SCC screening bases for the CSB flexures welds (CE-ID Item 7). | **Limitation 3 (L-03)**:<br><br>The scope of the WCAP-17096-NP, Revision 3, is limited in that it no longer includes or defines the acceptance criteria or data analysis criteria for performing the fatigue and SCC screening analyses of CE-design core barrel flexure welds or fatigue screening analyses of CE-design core support plates and upper fuel alignment plates. Consistent with the guidelines in MRP-227, Revision 1-A, a licensee using the subject TR should address the acceptance criteria and data analysis criteria for performing the component-specific screening analyses of CE-design core barrel flexure welds, core support plates, and upper fuel alignment plates on a unit-specific or site-specific case-by-case basis. | Rev.3 Section 3.6<br><br>Rev. 3 Appendix C, CE-ID Item 7,<br><br>Rev. 3 Appendix C, CE-ID Item 8<br><br>Rev. 3 Appendix C, CE-ID Item 9 | 3.4.8 |



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|---|---|---|---|---|---|
| Numerical Identifier for Limitations and Conditions | | | Limitations and Conditions | Rev. 3 Topical Report (TR) & Safety Eval. (SE) Sections | |
| Number | Topic / RAI | Technical Basis | | TR | SE |
| Limitation L-04 | In Revision 3 of the TR, the PWROG's bases for defining relevant conditions for B&W-design Primary category and Expansion category RVI components that are within the scope of EPRI MRP VT-3 visual examinations methods.<br>/<br>RAI-12 | In the PWROG's response to RAI-12, the referenced B&W-ID items include general provisions for identification of: (1) "what visual examination wear indications are considered rejectable and would require additional examination and evaluation," and (2) "any additional examination results that are anticipated" and the "general acceptance criteria for these additional items are expected to be in the VT-3 visual examination field of vision." However, the PWROG has confirmed that these statements do not fully define the acceptance and data analysis criteria for the components in these B&W-ID item assessments. | **Limitation 4 (L-04)**:<br><br>The scope of the WCAP-17096-NP, Revision 3, is limited in that it does not define any acceptance criteria or data analysis criteria for evaluating relevant conditions in B&W-design RVI components that are inspected using VT-3 visual inspection methods. Consistent with the guidelines in MRP-227, Revision 1-A, the acceptance criteria and data analysis criteria for evaluating relevant conditions in B&W-design Primary or Expansion category components that are subject to VT-3 inspections are to be addressed on a component-specific case-by-case basis by a licensee using WCAP-17096-NP, Revision 3. | Rev. 3 of the TR Appendix A,<br><br>B&W ID-Items A.1.1, A1.2, A.1.3, A.1.6, A.1.7, A.1.9, A.1.10, A.1.11, A.1.12, A.1.14, A.2.1, A.2.3, A.2.4, A.2.10, A.2.12 | 3.4.11 |



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|---|---|---|---|---|---|
| Numerical Identifier for Limitations and Conditions | | | Limitations and Conditions | Topical Report (TR) & Safety Eval. (SE) Sections | |
| Number | Topic / RAI | Technical Basis | | TR | SE |
| Condition C-01 | Selection and justification of CGRs for stainless steel metal and weld components.<br><br>NRC staff's C-01 corresponds with the PWROG's conditional statements in Revision 3 of the TR (regarding the use of bounding CGRs), with additional considerations when the licensee decides to use an alternate CGR basis from that defined by the PWROG. | The NRC staff's condition reflects the PWROG's conditional statements in its response to RAI-04 for the subject TR regarding the use of bounding CGRs for stainless steel components and welds (Reference 3).<br><br>RAI-04 applied to 16 high fluence components. However, in the response to RAI-04, the PWROG established bounding CGRs for stainless steel welds with neutron fluence exposures shown in Table 2 of this Final SE. Therefore, Condition 1 is applicable to all 23 components (i.e., 2 B&W, 14 CE, and 7 Westinghouse) that use the flaw tolerance evaluation method.<br><br>The CGRs were based on the CGR models in the EPRI report cited as *Reference 46* in the subject TR, which were also used to develop bounding CGRs in ASME Code Case N-889. | **Condition 1 (C-01)**:<br><br>For stainless steel metal and weld components in the specified neutron fluence exposure ranges as shown in Table 2 in Section 3.3.4 of this SE, the PWROG has indicated that the CGRs are based on the CGR criteria fluence ranges specified in ASME Code Case N-889, as established by the EPRI models in the EPRI Report cited as *Reference 46* in WCAP-17096-NP, Revision 3. The NRC has endorsed Code Case N-889 in Regulatory Guide 1.147, Revision 20, with conditions for the specified neutron fluence exposure ranges. Therefore, if the CGR models in *Reference 46* of WCAP-17096-NP, Revision 3, are followed by a licensee using WCAP-17096-NP, Revision 3, then the CGR models are subject to the NRC conditions on Code Case N-889, as defined in Table 2 of Regulatory Guide 1.147, Revision 20. | Rev. 3 of the TR Section 2.2<br><br>Appendix A, B&W-ID Items A.1.9 and A.2.11<br><br>Appendix C, CE-ID Items 2, 2.1, 3, 3.1, 5, 5.1, 5.2, 5.3 5.4, 6, 6.1, 6.2, 7, and 11<br><br>Appendix E, W-ID Items 3, 3.1, 3.2, 3.3, 4, 4.2 and 4.3 | 3.3.4 and 3.3.7 |



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|---|---|---|---|---|---|
| Numerical Identifier for Limitations and Conditions | | | Limitations and Conditions | Topical Report (TR) & Safety Eval. (SE) Sections | |
| Number | Topic / RAI | Technical Basis | | TR | SE |
| Condition C-02 | Use of predictive bolting pattern model or methodology bases for bolts in bolted assemblies. / RAI-07, Part (c), and RAI-13, Parts (a), and (b) | In the NRC staff's resolution of RAI-13, Parts (a), and (b), in this SE, the staff established its SE evaluation position that predictive bolting pattern model or methods may be used to establish the reinspection interval for a specified bolted assembly type or even as a basis for justifying further service of a bolted assembly when evidence of degradation in a number of bolts in the assembly and the methodology can support a finding that there would be a sufficient number of bolts, as based on the ability of the model to generate a predictive set of future bolting pattern outputs starting from an initial degraded bolting pattern (as established from the last inspections performed on the assembly). | **Condition 2 (C-02)**:<br><br>For bolted assembly analysis methods that use predictive, probabilistic ABPA models, the models and methods should appropriately account for limits on the number of allowable bolts with presumed failed conditions, geometric bolt failure considerations (i.e., bolt clustering considerations), and bolting analysis reliability considerations (i.e., addressing 95% reliability confidence limits). These models and methods should be appropriately documented and justified in a site-specific or owner-defined record. | WCAP-17096-NP, Revision 3, Section 2.3<br><br>WCAP-17096-NP, Revision 3, Section 3.3<br><br>WCAP-17096-NP, Revision 3, Appendices A, C, & E: Applicable B&W-ID, CE-ID and W-ID items for bolted assemblies in the subject TR appendix that rely or may rely on acceptable bolting pattern bases for structural integrity of the assembly. | 3.4.5 and 3.4.6 |



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|---|---|---|---|---|---|
| Numerical Identifier for Limitations and Conditions | | | Limitations and Conditions | Topical Report (TR) & Safety Eval. (SE) Sections | |
| Number | Topic / RAI | Technical Basis | | TR | SE |
| Condition C-03 | Accuracy and validation of lower bound LEFM fracture values reported in Revision 3 of TR and the validity and accuracy of neutron ranges reported for the specified fracture toughness values.<br>/<br>RAI-10 | As has been discussed in Section 3.4.10 of this SE, some neutron fluence ranges reported for specified lower bound fracture toughness values cited in Rev. 3 of the TR may no longer be valid per EPRI's Part 21 notification on the lower bound fracture toughness values, as reported in EPRI BWRVIP Letter No. 2021-030 (Reference 42). Based on the information in EPRI BWRVIP Letter No. 2021-020, there may be some uncertainty with the accuracy of the fluence ranges reported for specified lower bound fracture toughness values that were reported in the fracture toughness tables of Appendices C and E in Revision 3 of the TR. | **Condition 3 (C-03)**:<br><br>Until such time that the EPRI or PWROG can resolve the EPRI BWRVIP Letter No. 2021-020 Part 21 issue on neutron fluence ranges associated with specified lower bound fracture toughness values reported in an updated version of MRP-210 or MRP-211 (or alternate industry record), licensees will need to justify any lower bound fracture toughness values used as inputs in component-specific flaw tolerance analyses. The selection of the applicable lower bound fracture toughness value(s) should be reported and justified in a site-specific record and reconciled with any updated industry reports or records used to correlate lower bound fracture toughness value(s) and the most recent neutron fluence range(s) reported for the specified fracture toughness value(s). | WCAP-17096-NP, Revision 3, Appendix C, CE-ID Items 2, 2.1, 3, 3.1, 5, 5.1, 5.2, 5.3, 5.4, 6, 6.1, 6.2, 7, & 11,<br><br>and<br><br>WCAP-17096-NP, Revision 3, Appendix E, W-ID Items 3, 3.1, 3.2, 3.3, 4, 4.2, & 4.3 | 3.4.10 |



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# Safety Evaluation, Attachment B

## Resolution of Comments for the

## U.S. Nuclear Regulatory Commission's Draft Safety Evaluation of

## Pressurized Water Reactors Owners Group Topical Report No. WCAP-17096, Revision 3

On September 30, 2022, the staff submitted a copy of the draft safety evaluation (DSE) (Agencywide Documents Access and Management System (ADAMS) Accession No. ML22181B009) for Pressurized Water Reactor Owners Group (PWROG) Topical Report (TR) No. WCAP-17096, Revision 3 , for commentary by members of the PWROG (refer to the staff's transmittal email at ADAMS Accession No. ML22266A280).

The staff received a total of 73 comments regarding the DSE for the subject TR, including: (1) a total of four comments from Framatome as the PWROG's participating Nuclear Steam Supply System (NSSS) vendor for Babcock and Wilcox (B&W)-designed pressurized water reactors (PWRs), (2) a total of 68 comments from Westinghouse Electric Company as the PWROG's participating NSSS vendor for Combustion Engineering (CE)-designed and Westinghouse-designed PWRs, and (3) an additional email comment from the PWROG (Mr. J. Andrachek) on the appropriateness of citing the *"> 6.7x10$^{21}$" n/cm$^2$ (E> 1.0 MeV)*" fluence exposure value and the "> *than 10 dpa*" projected dose exposure value as proprietary contents in the DSE for the subject TR. The comments received from Westinghouse Electric Company and Framatome Corporation are in ADAMS package Accession No. ML23009A588. The comment received via email from the PWROG is located at ADAMS Accession No. ML23173A090 (Non-Public).

The comments and the staff's basis for addressing and resolving comments received on the DSE for the subject TR are provided in the table that follows.

Note: The first four comments were from Framatome. The remaining comments were from Westinghouse except for the final comment via email from the PWROG.

Attachment B



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Resolution of Comments Received on the DSE for WCAP-17096, Revision 3

| Comment No. | Text Location in DSE | | Comment Type (Clarification, Editorial, Basis, Accuracy, Proprietary) | Comment, Including Applicable PWROG Suggested Resolution or Revision | NRC Staff Basis for Responding to and Resolving Specified Comment Received |
|---|---|---|---|---|---|
| | Page(s) (See Note 1) | Line(s) (See Note 1) | | | |
| **Comments Received from Framatome Corporation** | | | | | |
| 1 | 38 – 40 | 33 on Page 38 – 1 on Page 40 | Accuracy and Basis | Comment:<br><br>"The primary purpose of the TR is to provide evaluation methodologies to address relevant conditions. However, per Section 5.2 of the TR, for the B&W units, the TR also "indicates actions that might be used to define nondestructive examination (NDE) acceptance standards and actions that could support analytical evaluations."<br><br>The NRC states the TR does not define acceptance criteria or data analysis criteria for evaluating relevant conditions for the RV internals components inspected using VT-3 visual inspection methods. It is unclear why the NRC is stating this. Each component item has a stated analytical methodology to be used for the acceptance criteria. For example, the methodology for evaluation for the CRGT spacer castings (A.1.2 in the TR) is provided on pages A-5 and A-6 (i.e., drop time evaluation or reactivity analysis).<br><br>Directly following this is the statement, "The general methodology to be used for acceptance criteria for these component items will be development of an NDE inspection standard that contains examples of acceptable and unacceptable visual indications and mockups for the VT-3 inspection of fractured spacers or missing screws." This is unrelated to the evaluation methodology. This statement is addressing NDE acceptance standards (per Section 5.2 of the TR)." | The staff did not find Framatome Comment 1 to be valid or the recommended action to be acceptable for implementation. Specifically, the staff's placement of Limitation L-04 in the DSE for WCAP-17096, Revision 3 applies to those B&W-design Primary and Expansion category components that are inspected by VT-3 visual inspection methods for conditions induced by aging effects in the components. The VT-3 visual methods look for gross conditions occurring in the components (e.g., gross indications of wear, loose parts, missing or severed parts, or distortion).<br><br>In its response to RAI 12 (ADAMS Accession No. ML22018A071), Framatome only made the following general statements regarding the acceptance standards for VT-3 detected relevant conditions and analytical efforts used to disposition the relevant conditions:<br><br>*"The general methodology to be used for VT-3 (a general condition monitoring visual examination) acceptance criteria for these component items will be development of an NDE inspection standard that contains explanations of acceptable and unacceptable visual indications . . . Analytical efforts may be performed on a unit-specific basis due to the different interference. The NE inspection standard could be developed generically."* |



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| Comment No. | Text Location in DSE | | Comment Type (Clarification, Editorial, Basis, Accuracy, Proprietary) | Comment, Including Applicable PWROG Suggested Resolution or Revision | NRC Staff Basis for Responding to and Resolving Specified Comment Received |
|---|---|---|---|---|---|
| | Page(s) (See Note 1) | Line(s) (See Note 1) | | | |
| 1 (Continued) | | | | Suggested Revision:<br><br>"Section 3.4.11 of the SE should be revised to state that the Staff accepts the response to RAI 12. Consequently, Limitation L-04 should be removed."<br><br>Staff Observation on Comment 1:<br><br>The potential relevancy of this comment should really be applied to all of Final SE Section 3.4.11. | Thus, even after the response to RAI 12 was received by the staff, the component-specific ID item assessments for the B&W-design components subject to VT-3 in Appendix A of the subject TR did not provide sufficient technical details on defining specific acceptance criteria or data analysis criteria for analyses that might be performed on B&W-design components with non-conforming relevant conditions. That is the reason why the response to RAI 12 was not fully accepted by the staff and why Limitation L-04 was created and issued in the *draft SE for commentary*.<br><br>As stated in Limitation L-04, the licensee should ensure that the non-conforming condition detected by the VT-3 visual examination of the component is appropriately entered, dispositioned, and justified in the licensee's corrective actions program. |
| 2 | 29 | 47 - 50 | Clarification | Comment:<br><br>"The DSE states: *The NRC staff finds that for W-design plants, the specification in Revision 3 of the TR Appendix E, W-ID Item 6 to submit the plant-specific application of the predictive model methodology defined in W-ID Item 6 is an acceptable way for making the details of the predictive methodology available to the NRC.*<br><br>Per TR Appendix E, W-ID Item 6, submittal to the NRC is for information and is only required if 1) MRP 2017-009 indicates that the subsequent inspection interval is not to exceed 6 years | The staff agrees that the Framatome Comment 2 recommendation is valid for implementation. Specifically, the comment applies to W-ID Item 6 in Appendix E of the subject TR for Westinghouse-design BF bolts and bolt locking devices. The updated W-ID Item 6 for the Westinghouse-design BF bolts and bolt locking devices includes a new analysis submittal statement that was added to W-ID Item 6 in resolution of Condition Item 3 for Group 3 components (which included the W-ID Item 6 Westinghouse-design BF bolts, |



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| 2 (Continued) | | | | and/or 2) there is an evaluation to lengthen the determined inspection interval or to exceed the maximum inspection interval recommended in MRP 2017-009."<br><br>Suggested Revision:<br><br>The following sentence should be added to the paragraph at the bottom of Page 27 of the Final DSE.<br><br>"Per TR Appendix E, W-ID Item 6, submittal to the NRC is for information and is only required if 1) MRP 2017-009 indicates that the subsequent inspection interval is not to exceed 6 years and/or 2) there is an evaluation to lengthen the determined inspection interval or to exceed the maximum inspection interval recommended in MRP 2017-009." | as referenced in Section 2.2 of the subject TR).<br><br>As based on Section 2.2 of the subject TR, W-ID Item 6 in Appendix E of the subject TR now calls any analysis used to extend the inspection interval of the Westinghouse-design BF bolts to be submitted to the staff. Therefore, the staff finds the comment to be accurate and the staff will implement and will add the referenced sentence identified in Framatome Comment 2 to Section 3.4.6 of this Final SE for the subject TR with minor editorial changes. |
| 3 | 30 | 8 - 12 | Editorial | Comment:<br><br>"This is the sentence that is the subject of this comment: *This would apply even if the bolting analysis methodology was approved by NRC (e.g., the PWROG's referencing of WCAP-15029-P-A, Revision 1 as an ABPA methodology in Revision 3 of the TR Appendix E, W-ID Items 6 and 6.1 for W-design BFBs and barrel-to-former bolts) or if Revision 3 of the TR precluded the plant-specific bolted assembly and bolting analysis from being performed using an unapproved methodology.*<br><br>Previous sentences on Page 28 state the NRC staff would expect the bolting evaluation methods to be placed into the NRC licensee's corrective action program and to be fully defined, discussed, and justified in a site- | The staff acknowledges the comment. Upon further review, the staff's perspective is that the referenced sentence still provides an important clarification, which is that the applicable analysis should be placed into the licensee's corrective action program regardless of whether the analysis methodology was staff approved. Thus, the staff left the referenced sentence in place in Section 3.4.6 of this Final SE for the subject TR. |



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| 3 (Continued) | | | | specific document/record. Therefore, the italicized sentence is not needed."<br><br>Suggested Revision:<br><br>"Remove the italicized sentence." | |
| 4 | A-7 and A-8 | C-02 | Basis | Comment:<br><br>"All aspects of this condition are already present in the TR (A.1.8 and W-ID: 6). It is unclear why this condition is necessary."<br><br>Suggested Revision:<br><br>"Remove Condition C-02." | The stated Condition C-02 applies to the licensee-specific analytical corrective action bases for dispositioning flaw indications in B&W and Westinghouse and CE design bolting other than baffle-former (BF) bolts or core shroud bolting.<br><br>The objective of Condition C-02 is to ensure that the analysis or alternate corrective action basis (e.g., component-specific repair or replacement) used to disposition a non-conforming condition in the bolting type or bolted assembly will be entered into, dispositioned, and justified in the licensee's correction actions program. Condition C-02 remains as a Condition listed in Table A-1 of the Final SE for the subject TR. |
| **Comments Received from Westinghouse Electric Company** | | | | | |
| 1 | 9 | 37 | Accuracy | Comment (including suggested revision):<br><br>The word "methods" is missing from the quoted statement (i.e., should be "… NRC approved evaluation methods…"). | The staff agrees that the comment is valid. The staff will make the change (Final SE Section 3.2) to include the word "methods." |
| 2 | 14 | 29 – 30 | Clarification | Comment (including suggested revision):<br><br>The quoted statement beginning on Line 34 is from TR R3 (not revised by RAI-04). Therefore, | The staff agrees that the comment is valid. The staff will incorporate the change recommended in Comment 2 |



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| 2 (Continued) | | | | the statement, "In PWROG's response to RAI-04 (See Reference 25), it stated that Section 2.2 of Revision 3 of the TR was revised to include the following additional action…" should be changed to "Section 2.2 of Revision of the TR includes the following additional action…" | (located within Section 3.3.4 of this Final SE). |
| 3 | 15 | 5 - 7 | Editorial | Comment (including suggested revision):<br><br>Delete extraneous figure title, "Figure 2 – Neutron Fluence Values." | The staff agrees that the comment is valid. The staff adjusted the part of the sentence in this Final SE in accordance with the editorial comment and changed the term "Figure" to "Table." |
| 4 | 15 | 13 | Editorial | Comment (including suggested revision):<br><br>Capitalize "Code Case." | The staff agrees that the comment is valid. The staff capitalized the first letters of the words "Code Case" in this Final SE. |
| 5 | 15 | Footnote 3 | Editorial | Comment (including suggested revision):<br><br>Change "…cracking growth evaluations…" to "crack growth evaluation." | The staff agrees that the comment is valid. The staff adjusted Footnote 3 accordingly in this Final SE. |
| 6 | 16 | 26 - 27 | Clarification | Comment (including suggested revision):<br><br>Add footnote to "CE-design core support column bolts in CE design with bolted core shroud (CS) assemblies" to clarify that this component has been removed from TR R3 per the response to RAI 7(a). | The staff agrees that the comment is valid considering the change in the licensing status of the operating licenses for Ft. Calhoun and Palisades PWR units to permanently defueled technical specification requirements. The modified version of the requested footnote on the component-specific bullet for the CE-design core support column bolts (DSE Page 16, Lines 26 - 27) was added as new Footnote 4 to this Final SE. |
| 7 | 19 | 33 – 34 | Clarification | Comment (including suggested revision):<br><br>Add "(CE-ID 2.1)." | The staff understands the rationale for the comment; however, for SE bullets involving specified Primary or Expansion components, the assigned |



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| 7 (Continued) | | | | | NRC project manager made an editorial decision for the SE format not to include the applicable component ID item number for the specified component type in the applicable DSE bullet. So instead of the adding (“CE-ID Item 2.1)” to the applicable bullet for the core shroud remaining axial welds, the staff added the words “(CE-design units with welded core shrouds assembled in two vertical sections)” to the component-specific bullet item within Section 3.3.8 of this Final SE. This is the welded type of CE-design core shroud that is associated with CE-ID Item 2.1. |
| 8 | 19 | 43 – 44 | Clarification | Comment (including suggested revision):<br><br>Add “(CE-ID 3.1).” | The staff understands the rationale for the comment; however, for bullets involving specified Primary or Expansion components, the assigned NRC project manager made an editorial decision for the SE format not to include the applicable component ID item number for the specified component type in the applicable bullet. Instead of adding “CE-ID Item 3.1” to the applicable bullet for the core shroud remaining axial welds, the staff added the words “(CE-design units with welded core shrouds assembled with full height shroud plates)” to the component-specific bullet within Section 3.3.8 of the Final SE. This is the welded type of CE-design core shroud that is associated with CE-ID Item 3.1. |
| 9 | 23 | 14 | Accuracy | Comment (including suggested revision): | The staff agrees that the comment is valid since the indented sentence involves quoted material from the |



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| | | | | Spell out OE as, “operating experience” as done in the quoted sentence. | applicable referenced document. The staff adjusted the indented, italicized sentence located within Section 3.4.1 of this Final SE. |
| 10 | 24 | 40 | Clarification | Comment (including suggested revision):<br><br>Correct “Article “WB-3000” to “Article IWB-3000.” | The staff agrees that the comment is valid. The staff corrected the typographical error within Section 3.4.3 of this Final SE. |
| 11 | 25 | 8 | Editorial | Comment (including suggested revision):<br><br>Correct document number of “MRP-2018-022” to “MRP 2018-022.” | The staff agrees that the comment is valid. EPRI does not include hyphenation between “MRP” and the listed year in these types of EPRI letter references. Thus, it should be in reference to “MRP 2018-022” for accuracy. The staff made the editorial change to within this Final SE. |
| 12 | 25 | 14 | Clarification | Comment (including suggested revision):<br><br>Reference 33 in the DSE is an erratum to correct a typographical error in WCAP-15029-P-A/WCAP-15030-NP-A. Based on the context, Reference 33 should be updated to the submittal of WCAP-15029-P-A/WCAP-15030-NP-A to the Staff.<br><br>Staff Comment and Observation:<br><br>The comment associated with Comment 12 is actually in reference to the manner the Reference 33 record (as referenced on DSE Page 25, Line 14) is worded in DSE Section 7.0 – see related Westinghouse Comment 42. Per Comment 42, the PWROG indicated that Reference 33 should be citing: Westinghouse Reports, WCAP-15029-P-A, Rev. 0 and WCAP-15030-NP-A, Rev. 0, “Westinghouse | The staff partially agrees that the comment is valid. Westinghouse is correct that Reference 33 in DSE Section 7.0 should actually be referencing the submittal of the WCAP-15029-P-A/WCAP-15030-NP-A reports; but the staff’s perspective is that Reference 33 should be citing the latest staff-approved version of the Proprietary report (WCAP-15029-P-A, Revision 1), which was submitted in ADAMS Accession Package No. ML18306A583 and transmitted in PWROG Letter OG-18-226, dated October 18, 2018.<br><br>The staff amended Final SE Section 7.0, Reference 33 accordingly per Comment 12 and the corresponding comment in Westinghouse Comment |



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| 12 (Continued) | | | | Methodology for Evaluating the Acceptability of Baffle-Former-Barrel Bolting Distributions under Faulted Load Conditions," December 1998 / April 1999 (ADAMS Accession Pkg. No. ML20206G308). | 42. A footnote is included in this Final SE for Reference 33 regarding the errata for the report. |
| 13 | 25 | 18 | Clarification | Comment (including suggested revision):<br><br>Reference 38 is the SE for WCAP-15029 which, based upon the context of the sentence, is not the correct reference. This sentence should be updated to refer to a new reference (that is not currently included in Section 7.0 of the DSE): NRC Letter, "Office of Nuclear Reactor Regulation Staff Assessment of Electric Power Research Institute NEI 03-08, Revision 2, "NEEDED" Interim Guidance Regarding Baffle-Former Bolt Inspections in Westinghouse Design Pressurized Water Reactors" November 20, 2017 (ADAMS Accession No. ML17310A861).<br><br>Staff Comment and Observation:<br><br>Westinghouse Electric Company's comment associated with Comment 18 is in regard to the citation of Reference 38 on DSE Page 25, Line 18 (as linked to the staff's SE for the interim baffle-to-former bolts (BFBs) when compared to the referencing of Reference 38 in other portions of DSE Section 3.4.4 which cites the staff SE for WCAP-15029-P-A (Revision 0). | The staff agrees that the comment is valid. Westinghouse is correct that the statement on Page 25, Line 18, of the DSE should be referencing a new Section 7.0 reference for the staff's SE on the Interim BFB Guidance dated November 20, 2017 (ADAMS Accession No. ML17310A861) and not Reference 38. However, the staff's acceptance of Comment 44 also had some bearing on how the staff addressed and incorporated the Final SE changes recommended in both Westinghouse Comments 13 and 44.<br><br>Specifically, Comment 44 requests deletion of Section 7.0 Reference 36 due to its redundancy with Section 7.0 Reference 23, and Comment 13 recommends creation of a new reference for the staff's Agency Record used for acceptance of the industry's Interim BFB Guidance. Thus, the staff addressed both comment recommendations by amending the Final SE reference of Reference 38 to Reference 36, and by amending Reference 36 in this Final SE Section 7.0 to be the new Final SE Section 7.0 reference for the staff's Agency Record assessment on the Interim BFB Guidance. Thus, the Final SE Section 7.0, Reference 36 was not |



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| 13 (Continued) | | | | | deleted, but was amended as follows to achieve the same objective of the combined recommended actions in Comments 13 and 44:<br><br>*NRC Agency Record providing "Office of Nuclear Reactor Regulation Staff Assessment of Electric Power Research Institute NEI 03-08, Revision 2, "NEEDED" Interim Guidance Regarding Baffle-Former Bolt Inspections in Westinghouse Design Pressurized Water Reactors," November 20, 2017 (ADAMS Accession No. ML17310A861)."* |
| 14 | 26 | 24 | Accuracy | Comment (including suggested revision):<br><br>The acronym is missing from the quoted statement. "Westinghouse Owners Group" should be "Westinghouse Owners Group [WOG]." | The staff agrees that the comment is valid. The staff added the missing acronym to Section 3.4.5 of this Final SE. |
| 15 | 26 | 33 – 34 | Clarification | Comment (including suggested revision):<br><br>The PWROG is not aware of a separate transmittal for RAI-07. The RAIs on TR R3 were transmitted in Reference 23 dated September 21, 2022. Therefore, the date should be changed from "September 24, 2020" to "September 21, 2020," and the citation to "(Reference 36)" should be changed to "(Reference 23)." See related Comments 21 and 44. | The staff agrees that Comment 13 is valid but observes that the validity and the needed resolution of Comment 44 requested activity is also relevant to the needed resolution basis for resolving Comment 15.<br><br>The staff amended the date reference in this Final SE to cite it as "September 21, 2020." The hyperlinks for the reference section were removed. |
| 16 | 26 | 37 – 38 | Clarification | Comment (including suggested revision):<br><br>Reference 37 is the Westinghouse predictive model methodology presentation dated | The staff agrees that the comment is valid. The hyperlinks to the reference section were removed. |



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| 16 (Continued) | | | | September 18, 2019. It seems based on the context that this should be a reference to WCAP-15029-P-A. Therefore, "(Reference 37)" should be "(Reference 33)" per the previous update to Reference 33 provided in Comments 12 and 42. | |
| 17 | 27 | 12 – 13 | Clarification | Comment (including suggested revision):<br><br>This statement appears to be referring to submittal of WCAP-17096-NP, Revision 3 to the staff rather than issuance of WCAP-15029-P-A, Revision 1. Therefore, "…prior to the PWROGs issuance of the NRC-approved ABPA methodology in WCAP-15029-P-A, Revision 1," should be updated to, "…prior to the PWROGs submittal of WCAP-17096-NP, Revision 3 to the Staff for approval in July 2019…" | The staff agrees that the comment is valid but proposes an alternate solution to the comment. Westinghouse is referring to the clause in Lines 12 and 13 of the sentence currently on DSE Page 27, lines 10 – 15, which states: "*although WCAP-18034-P was issued prior to the PWROG's issuance of the approved ABPA methodology in WCAP-15029-P-A, Revision 1. . .*" Upon further review, the staff finds that the clause starting with the word "*although*" is not necessary for the context of the referenced sentence; thus, the clause starting with the word "*although*" was deleted from the referenced sentence. |
| 18 | 27 | 46 | Accuracy | Comment (including suggested revision):<br><br>A correction is needed to the quoted statement. The wording, "…may have changes over two decades…" should be, "…may have been changed over the two decades…" | The staff agrees that the comment is valid and made the editorial change identified in the comment. |
| 19 | 27 | 50 | Clarification | Comment (including suggested revision):<br><br>Part C of RAI-07 deleted a sentence in Section 3.3 of the TR R3, not the entire section. Therefore, the statement, "…the complete deletion of Section 3.3 to Revision 3 of the TR…" should be updated to, "… the deletion of | The staff agrees that the comment is valid and incorporated the editorial change identified in the comment into the Final SE. |



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| 19 (Continued) | | | | the statement in section 3.3 of Revision 3 of the TR…" | |
| 20 | 28 | 2 | Clarification | Comment (including suggested revision):<br><br>The updated RAI response was dated January 14, 2022. Therefore, "…February 8, 2022…" should be changed to "…January 14, 2022…" | The staff agrees that the comment is valid with respect to a need of referencing a proper date for the PWROG record containing the response to RAI-07. The RAI response is contained in ADAMS Accession No. ML22039A144, but the date of the record is January 13, 2022, and not January 14, 2022 (which instead applies to the transmittal letter). So, the staff amended the reference date of the RAI-07 response to "*January 13, 2022*." |
| 21 | 29 | 11 - 12 | Clarification | Comment (including suggested revision):<br><br>All RAIs on TR R3 were issued via Reference 23, dated September 21, 2020. Therefore, "September 24, 2020" should be updated to "September 21, 2022", and "Reference 36" should be updated to "Reference 23." See related comments 15 and 44. | The staff agrees that the comment is valid. For the RAI-13 record referenced on DSE Page 29, Line 11, the staff amended the cited date from "*September 24, 2020*" to "*September 21, 2020.*"<br><br>For the reference item associated with the RAI-13 record referenced on DSE Page 29, Line 12, the staff amended the reference cited to Reference 23. |
| 22 | 29 | 41 – 42 | Clarification | Comment (including suggested revision):<br><br>Reference 40 is MRP 2019-009 (CB weld interim guidance). BWRVIP 2019-016 was cited in TR R3 as an example of what the NRC has previously accepted for submittal of probabilistic fracture mechanics. Recommend adding BWRVIP 2019-016 as an additional reference: EPRI Letter, BWRVIP 2019-016, "White Paper on Suggested Content for PFM Submittals to | The staff agrees that the comment is valid. Upon further review, the staff confirmed and amended this Final SE with a new reference within this Final SE Section 7.0 that corresponds to the BWRVIP 2019-016 record and that the hyperlinked reference item cited in this Final SE cites and hyperlinks to the Final SE Section 7.0 reference item for BWRVIP 2019-016. Specifically: |



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| 22 (Continued) | | | | the NRC," February 27, 2019. See related Comment 46.<br><br>Staff comment and observation:<br><br>The staff perceives the comment to be an "accuracy" based comment with what DSE Section 7.0, Reference 40 should be citing in the hyperlinked Reference 40 record or alternatively whether Line 42 on DSE Page 29 should be referencing a new hyperlinked reference item in SE Section 7.0 that corresponds to the BWRVIP 2019-016 record (and whether the Reference item listed on DSE Page 29, Line 42 should be cited and hyperlinked to the new reference item that will be added to Final SE Section 7.0 for the BWRVIP 2019-016 record). | "*47. EPRI Letter No. BWRVIP 2019-016, "White Paper on Suggested Content for PFM Submittals to the NRC," February 27, 2019 (ADAMS Accession No. ML19241A545).*"<br><br>Refer to the staff's resolution basis for the related comments in joint Comments 45 and 46. |
| 23 | 32 | 28 | Editorial | Comment (including suggested revision):<br><br>Correct document number "TB 19-5" to "TB-19-5." | The staff agrees that the comment is valid and made the associated editorial change to this Final SE to correct the typographical error in the TB-19-5 reference. |
| 24 | 35 | 9 | Editorial | Comment (including suggested revision):<br><br>Correct "CE ID" to "CE-ID." | The staff agrees that the comment is valid and made the associated editorial change to this Final SE; the "CE-ID" reference mentioned in Comment 24 is currently located to Page 35, Line 17 of the Final SE. |
| 25 | 37 | 2 and 4 | Clarification | Comment (including suggested revision):<br><br>Please add the CE-ID number "(CE-ID 2.1)," to this component to distinguish between CE-ID 3.1 on Line 4. | The staff agrees that the comment is valid but proposes an alternative solution for resolution of the comment based on the NRC PM's formatting decision not to include specified ID item references for component-specific bullet items. For the referenced "Core Shroud |



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| 25 (Continued) | | | | | Assembly (Welded) – Remaining Axial Welds" bullet referenced in Comment 25, the staff added the words "(CE-design units with welded core shrouds assembled in two vertical sections)", which correspond to the type of welded CE-design unit core shroud associated with the CE-ID Item 2.1 assessment in Appendix C of the subject TR.<br><br>Similarly, for the corresponding "Core Shroud Assembly (Welded) – Remaining Axial Welds" bullet, the staff deleted the words "-ID 3.1" from the bullet item and add the words "(CE-design units with welded core shrouds assembled with full height shroud plates)", which corresponds to the type of welded CE-design unit core shroud assessed in the CE-ID Item 3.1 assessment of Appendix C of the subject TR. |
| 26 | 37 | 5 - 9 | Clarification | Comment (including suggested revision):<br><br>Please add "CE" before these component names to distinguish these from Westinghouse components.<br><br>Staff Comment and Observation: The specified comment applies to the five (5) bullets for specified CE-design core support barrel (CSB) weld types (i.e., bullets for CE CSB Assembly UGWs, UAWs, MGWs, MAWs, and LAWs). | The staff agrees that Comment 26 for the specified CE-design core barrel weld components is valid. The staff added "CE" at the beginning of the specified CE-design component-specific bullet items. Additionally, the "Lower Support Structure, Deep Beams" referenced in DSE Page 37, Line 10 of the DSE are only applicable to CE units with welded core shrouds assembled with full height shroud plates. The staff added to this Final SE a parenthetical clarification to the CE component-specific bullet for the deep beams. |



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| | Page(s) (See Note 1) | Line(s) (See Note 1) | | | |
| 27 | 37 | 11 - 15 | Clarification | Comment (including suggested revision):<br><br>Please add "W" before these component names to distinguish from CE components.<br><br>Staff comment and observation:<br><br>The specified comment applies to the five (5) bullets for specified Westinghouse-design core barrel weld types (i.e., bullets for Westinghouse Core Barrel Assembly UGWs, UAWs, LGWs, MAWs, and LAWs). | The staff agrees that Comment 27 for the specified Westinghouse-design core barrel weld components is valid. The staff added "W" at the beginning of the specified Westinghouse-design component-specific bullet items in this Final SE. |
| 28 | 37 | 1 - 15 | Clarification | Comment (including suggested revision):<br><br>A total of 21 CE and Westinghouse component methodologies use a flaw tolerance evaluation with the fracture toughness limits that are potentially impacted by the Open Item; however, only 15 components are listed. The following six additional components should be included:<br><br>• CE CSB Assembly, UFW<br>• CE CSB Assembly, LGW<br>• CE Lower Support Structure, Lower Core Support Beams<br>• CE CSB Assembly, CSB Flexure Weld<br>• W Core Barrel Assembly, UFW<br>• W Core Barrel Assembly, LFW<br><br>See related Comment 53. | Staff agrees comment is valid. The staff added following component-specific bullet items to this Final SE:<br><br>• CE CSB Assembly, UFW<br>• CE CSB Assembly, LGW<br>• CE Lower Support Structure, Lower Core Support Beams<br>• CE CSB Assembly, CSB Flexure Weld<br>• W Core Barrel Assembly, UFW<br>• W Core Barrel Assembly, LFW<br><br>The new component-specific bullet for "CE lower support structure, lower core support beams" includes the following parenthetical clause: "*(All CE-design units except those with welded core* |



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| | Page(s) (See Note 1) | Line(s) (See Note 1) | | | |
| 28 (Continued) | | | | | *shrouds assembled with full height shroud plates)*". |
| 29 | 37 | 22 – 23 | Clarification | Comment (including suggested revision):<br><br>These values are from the Executive Summary of MRP-211, which was made public when this was submitted to the Staff. Therefore, these values do not need to be redacted, and the Safety Evaluation can be considered non-proprietary. The transmittal letter for MRP-211 is: Electric Power Research Institute (EPRI), MRP 2017-036, "Transmittal of Revision 1 to EPRI Technical Reports MRP-175 and MRP-211," December 18, 2017 (ML17361A187).<br><br>Staff comment and observation:<br><br>A similar comment was provided in a PWROG email to the staff. The staff has included that email-based comment at the end of this table. | Staff agrees with the comment. See the staff's resolution of the similar PWROG email comment that is covered by a line item at the end of this table. The specified "*6.7x10$^{21}$" n/cm$^2$ (E> 1.0 MeV)*" fluence exposure and "*10 dpa*" projected dose exposure values in question are now included in this Final SE. The values were downgraded and listed as non-proprietary contents for this Final SE on the subject TR. In addition, the Proprietary headers and footers were removed from this Final SE per the staff's acceptance of this comment and the analogous email comments on proprietary contents in the DSE. |
| 30 | 39 | 34 | Editorial | Comment (including suggested revision):<br><br>Correct "W--D" to "W-ID." | The staff agrees that the comment is valid and fixed the typographical error of the acronym in this Final SE. The acronym now reads as "W-ID." |
| 31 | 43 | Ref. 2 | Clarification | Comment (including suggested revision):<br><br>Accession number should be "ML21099A172." | The staff recognizes that the comment is accurate, but no changes of the referenced ADAMS Package number needed to be made in response to Comment 31 for Reference 2. Specifically, the staff entered the same Package records into two different packages in ADAMS: (1) one by the Package in ADAMS Accession No. ML21099A172, as indicated in Comment 31, and (2) again in the Package in ADAMS Accession |



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| 31 (Continued) | | | | | No. ML22187A141, as currently indicated in Reference 2 of Section 7.0 of this Final SE.<br><br>Thus, the citing of Package Item ML22187A141 in Reference 2 is accurate, and the Package has the same records as the package in ADAMS Package No. ML21099A172. |
| 32 | 43 | Ref. 3 | Editorial | Comment (including suggested revision):<br><br>The PA number "(PA-MSC-1567)" is missing from the document title. The title should be, "Transmittal of the Response to Request for Additional Information RAIs Associated with WCAP-17096, Revision 3, "Reactor Internals Acceptance Criteria Methodology and Data Requirements," (PA-MSC-1567)." | The staff recognizes that the comment is accurate and valid in that the cited title for Reference 3 omitted the "(PA-MSC-1567)" term in the record title. The staff added the "(PA-MSC-1567)" term into the title of Reference 3 to make it accurate with the title of the actual record in ADAMS Package No. ML22187A149. |
| 33 | 43 | Ref. 3 | Clarification | Comment (including suggested revision):<br><br>Accession number should be "ML22039A142." | The staff recognizes that the comment is accurate, but no changes of the referenced ADAMS Package number needed to be made in response to Comment 33 and Reference 3. Specifically, the staff entered the same Package records into two different packages in ADAMS: (1) one by the Package in ADAMS Accession No. ML22039A142, as indicated in Comment 33, and (2) again in the Package in ADAMS Accession No. ML22187A149, as currently indicated in Reference 3 of Section 7.0 of this Final SE.<br><br>Thus, the citing of Package Item ML22187A149 is accurate, and the Package has the same records as the |



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| 33 (Continued) | | | | | package in ADAMS Package Item No. ML22039A142. |
| 34 | 43 | Ref. 4 | Clarification | Comment (including suggested revision):<br><br>The document revision number is missing from the citation and the title is incorrect. Revise citation as follows:<br><br>Westinghouse Electric Company Topical Report WCAP-17096-NP, Revision 3, "Reactor Internals Acceptance Criteria Methodology and Data Requirements," July 2019 (Pkg ML22187A179). | The staff agrees the comment is accurate and valid. The staff made the stated editorial changes to Reference 4; however, the term "Non-Proprietary Class 3" did not need to be removed from the Reference 4 item. |
| 35 | 43 | Ref. 13 | Clarification | Comment (including suggested revision):<br><br>Accession number should be "ML16279A320." | The staff agrees the comment is accurate and valid and that ADAMS No. ML16279A320 is the proper reference for WCAP-17096-NP-A, Rev. 2 (Reference 13).<br><br>The staff confirmed that ADAMS No. ML16274A399 is the ADAMS reference for the NRC's NUREG-2191, Volume 2, "Generic Aging Lessons Learned for Subsequent License Renewal GALL-SLR) Report"; thus, the citing of ADAMS No. ML16274A399 in DSE Reference 13 is incorrect and the staff corrected the citation of "ML16279A320" as the applicable ADAMS number for the Agency record cited for Reference 13 within Section 7.0 of this Final SE. |
| 36 | 44 | Ref. 24 | Clarification | Comment (including suggested revision):<br><br>Document date should be "January 14, 2022." | The staff agrees the comment is accurate and valid. The staff corrected the referenced date for Reference 24 within Section 7.0 of this Final SE. |



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| 37 | 44 | Ref. 25 | Clarification | Comment (including suggested revision):<br><br>Document date should be "January 13, 2022." | The staff agrees the comment is accurate and valid. The staff corrected the referenced date for Reference 25 within Section 7.0 of this Final SE. |
| 38 | 44 | Ref. 27 | Editorial | Comment (including suggested revision):<br><br>Misspelling of word in title. Should be "…Austenitic Stainless Steel…" | The staff agrees the comment is accurate and valid. The staff corrected the misspelling of the word "Austenitic" in the title of the Agency record that is cited for Reference 27 within Section 7.0 of this Final SE. |
| 39 | 45 | Ref. 30 | Editorial | Comment (including suggested revision):<br><br>Location of publisher should be, "New York, NY." | The staff agrees the comment is accurate and valid. The staff corrected the location of the cited publisher of the Agency Record cited for Reference 30 within Section 7.0 of this Final SE. |
| 40 | 45 | Ref. 31 | Clarification | Comment (including suggested revision):<br><br>Letter number is missing from the citation. Please add "MRP 2016-021." | The staff agrees the comment is accurate and valid. The staff inserted the words "Letter No. MRP 2016-021" into the Agency Record cited for Reference 31 within Section 7.0 of this Final SE. |
| 41 | 45 | Ref. 32 | Clarification | Comment (including suggested revision):<br><br>The letter number is missing from the citation. Please add "MRP 2017-009." | The staff agrees the comment is accurate and valid. The staff inserted the words "Letter No. MRP 2017-009" into the Agency Record cited for Reference 32 within Section 7.0 of this Final SE. |
| 42 | 45 | Ref. 33 | Clarification | Comment (including suggested revision):<br><br>Reference 33 should be the original NRC-approved Westinghouse ABPA methodology. This reference should be updated to: Westinghouse Reports, WCAP-15029-P-A, Rev. 0 and WCAP-15030-NP-A, Rev. 0, | The staff partially agrees that the comment is valid. Westinghouse is correct that Reference 33 in DSE Section 7.0 should have referenced the submittal of the WCAP-15029-P-A/ WCAP-15030-NP-A reports, but it should be to the latest approved version |



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| 42 (Continued) | | | | "Westinghouse Methodology for Evaluating the Acceptability of Baffle-Former-Barrel Bolting Distributions under Faulted Load Conditions," December 1998 / April 1999 (Pkg. No. ML20206G308).<br><br>It is recommended that WCAP-15029-P-A, Rev. 0 include a footnote to explain that WCAP-15029-P-A, Rev. 1 is equivalent to Rev. 0 with the Safety Evaluation attached.<br><br>Staff Comment and Observation:<br><br>Westinghouse Electric Company's comment associated with Comment 42 is actually in reference to the manner the Reference 33 record is worded in DSE Section 7.0 (on SE page 43) – see Westinghouse related Comment 12. Per Comment 42, the PWROG indicates that Reference 33 should be citing: Westinghouse Reports, WCAP-15029-P-A, Rev. 0 and WCAP-15030-NP-A, Rev. 0, "Westinghouse Methodology for Evaluating the Acceptability of Baffle-Former-Barrel Bolting Distributions under Faulted Load Conditions," December 1998 / April 1999 (Pkg. No. ML20206G308). However, the updated staff-approved version of the proprietary report (WCAP-15029-P-A, Rev. 1) was submitted in PWROG Letter No. OG-18-226, Oct. 31, 2018. Thus, the staff is of the opinion Reference 33 (now on Final SE Page 45) should be citing the updated staff-approved version in WCAP-15029-P-A, Rev. 1, as included in OG-18-226. | of the Proprietary report (WCAP-15029-P-A, Rev. 1), which was transmitted in PWROG Letter OG-18-226, dated October 8, 2018, and included ADAMS Package No. ML18306A583.<br><br>The staff amended Reference 33 within Section 7.0 of this Final SE per the corresponding comments in Westinghouse Comments 12 and 42, but as modified by the staff's observations for Reference 33 regarding the ADAMS Package number for the latest staff-approved version of WCAP-15029-P-A (currently Revision 1). A footnote was included in the amended version of Reference 33 regarding the errata for the report.<br><br>Based on acceptance of the Comments 12 and 42, the staff confirmed the proper reference for Reference 33 is given in PWROG Letter No. OG-18-226, October 31, 2018 (ADAMS Package No. ML18306A583), which has the latest staff-approved versions of the WCAP-15029-P-A and WCAP-15030-NP-A reports (i.e., Revision 1 and Revision 0, respectively). |
| 43 | 45 | Ref. 35 | Editorial/Clarification | Comment (including suggested revision):<br><br>The accession number of the non-proprietary version needs to be corrected and should be | The staff agrees the comment is valid, and that a number of corrections to Reference 35 were necessary. Reference 35 should be the citation for |



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| 43 (Continued) | | | | indicated as "non-proprietary." Also, ML15223A269 cannot be located in ADAMS. The corrected accession number info is: "(ML080110645 – Non-Proprietary/Public) …" | the MRP-211, Revision 1 report, as cited and hyperlinked to Reference 35 on Page 36 of the DSE. But Page 34 improperly stated that Reference 35 is for the BWRVIP-100-A report. For the Final SE to be correct, Section 7.0 was corrected to give the Reference 35 citation as the cited record for the EPRI MRP-211 report (EPRI Report 3002010270).<br><br>The version of BWRVIP-100 cited in the DSE should be BWRVIP-100, Revision 1-A, which is Reference 41 within Section 7.0 of this Final SE. |
| 44 | 45 | Ref. 36 | Clarification | Comment (including suggested revision):<br><br>All of the RAIs for TR R3 were provided in Reference 23. This ML number cannot be found in ADAMS. This reference should be deleted. See related Comments 15 and 21. | See the staff's resolution of the suggested action in Westinghouse Comment 13, which includes the staff's basis for amending Reference 36 within Section 7.0 of this Final SE per the joint comment amendment actions recommended in Westinghouse Comments 13 and 44. |
| 45 and 46 | 46 | Ref. 40 | Editorial | Comments (including suggested revision):<br><br>Correct the document number from "MRP-2019-023" to "MRP 2019-023." BWRVIP 2019-016 is not associated with MRP 2019-023. MRP 2019-023 transmits letter MRP 2019-009. Therefore, "(BWRVIP 2019-016)" should be changed to "(MRP 2019-009)." | The staff agrees the comment is valid and Reference 46 within Section 7.0 of this Final SE was revised. However, BWRVIP 2019-016 is correct when taken in context with Comment 22 and the staff's basis for resolving Comment 22. The staff made the associated editorial change to Reference 47 within Section 7.0 of this Final SE. |
| 47 | 46 | Ref. 42 | Editorial | Comment (including suggested revision):<br><br>Correct the document title: change "BWR-VIP-100" should be changed to "BWRVIP-100." | The staff agrees the comments are valid. The staff made the editorial change to Reference 42 within Section 7.0 of this Final SE. |



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| 48 | 46 | Ref. 43 | Clarification | Comment (including suggested revision):<br><br>The document is a repeat of Reference 40 and can be deleted. | The staff agrees the comment is valid. The staff deleted Reference 43. This resulted in a shift of the Reference numbers up one for references coming after Reference 43 (e.g., Reference 44 became the new Reference 43, and Reference 45 became the new Reference 44, etc.) The referencing of those references in the body of the Final SE was adjusted accordingly. |
| 49 | 43 | Ref. 44 | Editorial | Comment (including suggested revision):<br><br>Correction to title, “CWAP-17096” should be “WCAP-17096.” | The staff agrees the comment is valid. The staff made the editorial change to Reference 44 of this Final SE which became the new Reference 43 after the Comment 48 change was implemented. |
| 50 | A-3 | L-03 | Clarification | Comment (including suggested revision):<br><br>“Rev. 3 Appendix C, CE-ID Item 8” is listed twice. One instance should be deleted to clarify the intent. | The staff agrees the comment is valid and deleted the extra reference of the “Ref. 3 Appendix C, CE-ID Item 8” in the last column entry of this Final SE Table A-1, Limitation L-03 line item. |
| 51 | A-5 | C-01 | Clarification | Comment (including suggested revision):<br><br>The following statement is included in the technical basis, “In the response to RAI-04, the PWROG established bounding CGRs for stainless steel welds with neutron fluence exposures shown in Figure 2 of this SE.”<br><br>RAI-04 was only related to high fluence components; therefore,16 high fluence components were identified in the response to RAI-04. However, since ASME Code Case N-889 applies to all fluence ranges in Figure 2, a total of 23 components are affected (2 B&W, 14 CE, and 7 Westinghouse). | The staff agrees the comment is valid. The following statement in the “Technical Basis” column entry in Table 1, Condition C-01 line item of the DSE:<br><br>“*In the response to RAI-04, the PWROG established bounding CGRs for stainless steel welds with neutron fluence exposures in the fluence ranges shown in Figure 2 of this SE.*”<br><br>Now reads as the following in the Final SE: |



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| 51 (Continued) | | | | It is proposed that this statement is updated as followed to clarify:<br><br>"RAI-04 applied to 16 high fluence components. However, in the response to RAI-04, the PWROG established bounding CGRs for stainless steel welds with neutron fluence exposures shown in Figure 2 of this SE. Therefore, Condition 1 is applicable to all 23 components (i.e., 2 B&W, 14 CE, and 7 Westinghouse) that use the flaw tolerance evaluation method." See Comment 52. | "*RAI-04 applied to 16 high fluence components. However, in the response to RAI-04, the PWROG established bounding CGRs for stainless steel welds with neutron fluence exposures shown in Table 2 of this SE. Therefore, Condition 1 is applicable to all 23 components (i.e., 2 B&W, 14 CE, and 7 Westinghouse) that use the flaw tolerance evaluation method.*" |
| 52 | A-5 | C-01 | Clarification | Comment (including suggested revision):<br><br>There are two typographical errors in the ID numbers of the 23 components. The B&W lower grid rib section is "A.2.11" rather than "A.2.3." Also, the CE Lower Support Structure is "CE-ID 11" rather than "CE-ID 1". | The staff agrees the comment is valid. The staff made the applicable editorial changes to the last column entry for the Condition C-01 line item in Table A-1 of this Final SE. Refer to the staff's resolution of Comment 51 as well. |
| 53 | A-7 | OI-01 | Clarification | Comment (including suggested revision):<br><br>The fracture toughness Open Item 1 applies to 21 components, which should be identified in Table OI-01. Therefore, CE-ID 5 (CSB Assembly – Upper Flange Weld) should be added. See related comment 28. | The staff agrees that the intent of the comment is valid. However, subsequent to receipt of Comment 53, the staff determined that it could not issue the final SE with an open item (OI-01). Accordingly, the staff converted OI-01 into a new Condition (new Condition C-03). Thus, the staff points out that the ID item referenced in Comment 53 as previously applying to Open Item OI-01 (deleted in this Final SE) now applies to new Condition C-03 in this Final SE. |
| 54 | 4 | Figure 1 | Editorial | Comment (including suggested revision):<br><br>Hyperlinks do not work. Correct hyperlinks as necessary. | Hyperlinks to the reference section were removed. |
| 55 | 4 | Ref. 19 in | Clarification | Comment (including suggested revision): | The staff agrees the comment is valid. The staff made the applicable editorial |



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| 55 (Continued) | | Figure 1 | | A correction is needed to the SLR-Interim Staff Guidance document title. “NRC’s SLR-Interim Staff Guiance-PWRVI-2021-01-“ should be “NRC’s SLR-Interim Staff Guidance-2021-01-PWRVI.” | changes the line items for Interim Staff Guidance No. SLR-ISG-2021-01-PWRVI and the Federal Register Notice for the SLR-ISG in Table 1 of this Final SE for the subject TR. |
| 56 | 5 | 38 | Editorial | Comment (including suggested revision):<br><br>Delete extraneous “an” from sentence to clarify: “NRC staff’s determination that the PWROG-specified an acceptance criteria methodology or data requirement basis for a given component ID item assessment in Appendix A, C, or E of Revision 3 of the TR was…” should be “NRC staff’s determination that the PWROG-specified acceptance criteria methodology or data requirement basis for a given component ID items assessment in Appendix A, C, or E of Revision 3 of the TR was…” | The staff agrees the comment is valid and deleted the extraneous “an” from the referenced bullet item. |
| 57 | 10 | 1 – 6 | Clarification | Comment (including suggested revision):<br><br>This paragraph discusses use of the WCAP-17096-NP acceptance criteria methodology and data requirements to “meet the acceptance criteria for the component item in Section 5 of MRP-227, Rev. 1-A.” This seems to be referring to the “Examination Acceptance Criteria” column in Table 5-1, 5-2, and 5-3. These Examination Acceptance Criteria are the specific relevant (or reportable) conditions used by an inspector when performing the MRP-227 inspections. The MRP-227, Section 5 Examination Acceptance criteria are not related to the WCAP-17096 methodologies, which are intended to create criteria for addressing the relevant conditions that are identified in inspections. Therefore, this paragraph discusses cases that are not relevant | The staff acknowledges the comment and agrees the comment appears to make a valid point. Consistent with the comment, the staff deleted the paragraph in this Final SE. |



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| 57 (Continued) | | | | and may add confusion to the Final SE, and it is requested that this paragraph be deleted. | |
| 58 | 11 | 15 – 19 | Clarification | Comment (including suggested revision):<br><br>A change in the inspection categorization would not impact the methodologies in WCAP-17096-NP, Rev. 3; however, a change in the type of inspection could possibly impact the method. Correct “inspection categorizations” to “type of inspection.” | The staff did not find the Westinghouse Comment 58 to be valid for implementation because the point of the limitation in Limitation L-01 applies to potential use of applicable component-specific ID item assessments for B&W, CE, and Westinghouse design Primary and Expansion components in Appendix A, C, or E of WCAP-17096, Revision 3 when compared to an updated list of B&W, CE, and Westinghouse design Primary and Expansion category components in an 80-year version of the MRP-227 report (e.g., a staff-approved version of MRP-227, Revision 2, which was submitted for staff review in early May of 2022).<br><br>Specifically, the list of Primary and Expansion category component ID item assessments in Appendices A, C, and E of the subject TR (WCAP-17096, Rev. 3) correspond to the set of Primary and Expansion category components as approved in the MRP-227, Revision 1-A; however, the Primary and Expansion category components addressed by the ID item assessments in Appendices A, C, and E of the subject TR do not perfectly correspond to the set of Primary and Expansion category components for B&W, CE, and Westinghouse components in the MRP-227, Revision 2 report, where there were some changes to the defined set of Primary and Expansion |



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| 58 (Continued) | | | | | category components in MRP-227, Revision 2 from the prior set of Primary and Expansion category components in MRP-227, Revision 1-A. Thus, the staff's statement on Page 11, Lines 15 – 19 of the DSE (as referenced in Westinghouse Comment 58) is appropriately being made in reference to changes in "inspection category" and not to changes in "inspection method." The sentence was left as written in this Final SE. |
| 59 | 16 | 28 | Clarification | Comment (including suggested revision):<br><br>"CE-designed core support' should be "CE-designed core support columns" | The staff agrees the comment is valid and made the editorial change of the referenced bullet item to list the applicable CE-design RVI component as "CE-design cores support columns." |
| 60 | 17 | 37 | Editorial | Comment (including suggested revision):<br><br>"W-designed CB middle axial welds" should be "W-designed CB MAWs" since MAWs was already defined above. | The staff agrees the comment is valid and made the editorial change of the referenced SE bullet in this Final SE. |
| 61 | 17 | 47 | Clarification | Comment (including suggested revision):<br><br>Correct "one year" to "next RFO". | The staff agrees the comment is valid. The staff made an editorial change within this Final SE that replaces the words "*no later than one year*" with the words "*at the next refueling outage (RFO)* . . ." |
| 62 | 19 | 31 | Editorial | Comment (including suggested revision):<br><br>This is the first item being listed and should be indented and bulletized like the other four items. | The staff agrees the comment is valid and has indented the bullet for the CE-design core shroud plate-to-former plate welds accordingly within this Final SE. |
| 63 | 30 | Footnote 7 | Editorial | Comment (including suggested revision): | The staff agrees the comment is valid and applies throughout the DSE in a manner that is more generic than |



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|---|---|---|---|---|---|
| | Page(s) (See Note 1) | Line(s) (See Note 1) | | | |
| 63 (Continued) | | | | "APB" should be corrected to "ABP" in the sentence "… for the assembly that would then be compared to the APB or set of ABPs," should be "… for the assembly would be compared to the ABP or set of ABPs."<br><br>Staff Comment and Observation:<br><br>The staff observes that the type of typographical error referenced in Westinghouse Comment 63 applies more generically than just the noted typographical error of the acronym "APB" listed in the referenced SE footnote. It applies to some existing typographical errors of the acronyms for approved bolting patterns (incorrectly listed as "APB") or approved bolting pattern analyses (incorrectly listed as "APBA") throughout the DSE for commentary, in which the acronyms need to be corrected to "ABP" or "ABPA". | implied in Westinghouse Comment 63. The staff searched the DSE for the term "APB" and globally changed that to "ABP" within this Final SE. The staff also searched the DSE for the term "APBA" and changed that globally to "ABPA" within this Final SE. |
| 64 | 33 | 1 – 2 | Clarification | Comment (including suggested revision):<br><br>The operating experience being referred to in "and with the exception of distortion recently reported in one of the clevis insert assemblies in a U.S. W-design PWR" was not considered "distortion" of the component. The clevis insert at that plant was displaced from its installed location. Discussion of that operating experience in the context of void swelling is misleading both due to the lack of "distortion" in the experience and due to the fact that the clevis insert assembly is located far from the core and receives very low radiation doses—orders of magnitude below the amount needed to cause void swelling distortion. It is recommended that "(and with the exception of distortion recently reported in one of the clevis | The staff acknowledges the comment may be valid. Upon further review, the staff has determined that the referenced parenthetical clause *"(and with the exception of distortion in one of the clevis insert assemblies in a U.S. design PWR)*" is not necessary for the referenced sentence on Page 33, Lines 1 – 2 of the DSE. The staff deleted the reference parenthetical clause *"(and with the exception of distortion in one of the clevis insert assemblies in a U.S. design PWR)*" from the scope of the sentence within this Final SE. |



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| Comment No. | Text Location in DSE | | Comment Type (Clarification, Editorial, Basis, Accuracy, Proprietary) | Comment, Including Applicable PWROG Suggested Resolution or Revision | NRC Staff Basis for Responding to and Resolving Specified Comment Received |
|---|---|---|---|---|---|
| | Page(s) (See Note 1) | Line(s) (See Note 1) | | | |
| 64 (Continued) | | | | insert assemblies in a U.S. W-design PWR)" be deleted. | |
| 65, 66, 67, and 68 | 38 and 39 | P38, Lines 41 and 46,<br><br>P39, Lines 2 and 8 | Clarification | Comment (including suggested revision) applies to Comments 65, 66, 67, and 68):<br><br>Correct "VT-3" to "EVT-1, ET, or UT."<br><br>Staff Comment and Observation:<br><br>Staff perceives the Westinghouse-recommended SE adjustment action in Westinghouse Comments 65 – 68 to be applicable to the staff's referencing of VT-3 inspection methods throughout DSE Section 3.4.11 and not just to the portions of DSE Section 3.4.11 that are located on Pages 38 and 39 of the "Draft SE for Commentary." Thus, the staff will respond to these comments based on the referencing of VT-3 inspection methods throughout Final SE Section 3.4.11. | The staff did not find these comments to be acceptable for implementation in Section 3.4.11 of the Final SE. Specifically, the use of "EVT-1, ET, or UT" as specified inspection techniques for replacing visual inspection methods in MRP-227 space typically applies to the EPRI MRP's expansion of the types of inspection methods that may be used for management of cracking in RVI structural seam weld types (e.g., those in the welds of welded core barrel or core support barrel assembly structures), as redefined and given in EPRI 80-Year (SLR-based) reports (e.g., MRP-227, Revision 2; MRP-231, Revision 4; MRP-232, Revision 2; or MRP 2018-022). But, the ID item assessments in Appendices A, C and E of the subject TR correspond to those Primary and Expansion category components in MRP-227, Revision 1-A, which is a 60-Year (initial LR-based) report.<br><br>In contrast, the staff observes that the referencing of VT-3 visual methods in Section 3.4.11 applies to use of VT-3 visual inspection methods for aging effects in specific Babcock and Wilcox (B&W)-design Primary or Expansion category components other than cracking, as linked to the VT-3 monitoring basis in Section 3.2 of the subject TR and a number of specified component-specific B&W ID item |



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| Comment No. | Text Location in DSE | | Comment Type (Clarification, Editorial, Basis, Accuracy, Proprietary) | Comment, Including Applicable PWROG Suggested Resolution or Revision | NRC Staff Basis for Responding to and Resolving Specified Comment Received |
|---|---|---|---|---|---|
| | Page(s) (See Note 1) | Line(s) (See Note 1) | | | |
| 65, 66, 67, and 68 (Continued) | | | | | assessments in Appendix A of the subject TR. For B&W-design Primary or Expansion category components associated with these B&W ID item assessments, the VT-3 methods are appropriate to monitor for conditions that may be indicative of gross conditions occurring in the specified B&W-design components (e.g., evidence of gross indications of wear, loose or missing parts, severed parts, etc.). Thus, the staff does not find recommended action for generically switching "VT-3" over to "EVT-1, ET, or UT" to be appropriate for the referencing those B&W-design ID item components in Appendix A of the subject TR that are the subject of the staff's evaluation in Section 3.4.11 and the VT-3 methodology described in Section 3.2 of the subject TR. |
| PWROG email Comment | 35 | 27 and 28 | Clarification<br><br>PWROG eMail comment (from Mr. J. Andrachek) on inclusion of proprietary lower bound fracture toughness values in the DSE. An analogous comment is given in Westinghouse Comment 29. | Summary of Comment (including suggested revision):<br><br>Proprietary values listed on DSE Page 35, Lines 27 and 28 can be downgraded to non-proprietary contents. The "> *6.7x10$^{21}$" n/cm$^2$ (E> 1.0 MeV)*" fluence exposure and "> *than 10 dpa*" projected dose exposure linked to the cited 34.6 ksi-√inch lower bound fracture toughness value in the DSE were included in the Executive Summary of the Proprietary MRP-211 Report (ADAMS ML17361A191); the Executive Summary is included in MRP-211, Rev. 1 as publicly available information.<br><br>The PWROG recommends downgrading the referenced "*6.7x10$^{21}$" n/cm$^2$ (E> 1.0 MeV)*" | The staff agrees that the comment is valid. The staff confirmed that the "*6.7x10$^{21}$" n/cm$^2$ (E> 1.0 MeV)*" fluence exposure and "*10 dpa*" projected dose exposure values linked to the cited 34.6 ksi-√inch lower bound fracture toughness value in the DSE were included in the Executive Summary of the EPRI MRP-211, Revision 1 report and that the Executive Summary in the report is given as publicly available information. The staff amended this Final SE to downgrade the cited "*6.7x10$^{21}$" n/cm$^2$ (E> 1.0 MeV)*" fluence exposure and "*10 dpa*" projected dose exposure values as non-proprietary values. In addition, the proprietary |



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| Comment No. | Text Location in DSE | | Comment Type (Clarification, Editorial, Basis, Accuracy, Proprietary) | Comment, Including Applicable PWROG Suggested Resolution or Revision | NRC Staff Basis for Responding to and Resolving Specified Comment Received |
|---|---|---|---|---|---|
| | Page(s) (See Note 1) | Line(s) (See Note 1) | | | |
| PWROG email Comment (Continued) | | | | fluence exposure and "*10 dpa*" projected dose exposure values to non-proprietary content status in the Final SE, which can result in issuance of the Final SE as a non-proprietary NRC Agency Record in ADAMS. | headers and footers contained in the "draft SE for commentary" were removed from the scope of this Final SE in accordance with this comment.<br><br>Consistent with the PWROG's perception, this results in the issuance of this Final SE for WCAP-17096, Rev. 3 as a non-proprietary agency record in ADAMS. |

Note: 1. The DSE pages and lines identified by Framatome Corporation or Westinghouse Electric Company in their comments did not account for the fact that the first two pages of MS Word file containing the "Draft SE Issued for Commentary" were taken up by the Table of Contents. The staff inserted the correct DSE (ML22181B009) page(s) and line number(s) within the Table of Attachment B of this Final SE.



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**PWR Owners Group**

**United States Member Participation* for PA-MSC-1567**

| Utility Member | Plant Site(s) | Participant | |
|---|---|---|---|
| | | Yes | No |
| Ameren Missouri | Callaway (W) | X | |
| American Electric Power | D.C. Cook 1 & 2 (W) | X | |
| Arizona Public Service | Palo Verde Unit 1, 2, & 3 (CE) | X | |
| Dominion Connecticut | Millstone 2 (CE) | X | |
| | Millstone 3 (W) | X | |
| Dominion VA | North Anna 1 & 2 (W) | X | |
| | Surry 1 & 2 (W) | X | |
| Duke Energy Carolinas | Catawba 1 & 2 (W) | X | |
| | McGuire 1 & 2 (W) | X | |
| | Oconee 1, 2, & 3 (B&W) | X | |
| Duke Energy Progress | Robinson 2 (W) | X | |
| | Shearon Harris (W) | X | |
| Entergy Palisades | Palisades (CE) | X | |
| Entergy Nuclear Northeast | Indian Point 2 & 3 (W) | X | |
| Entergy Operations South | Arkansas 1 (B&W) | X | |
| | Arkansas 2 (CE) | X | |
| | Waterford 3 (CE) | X | |
| Exelon Generation Co. LLC | Braidwood 1 & 2 (W) | X | |
| | Byron 1 & 2 (W) | X | |
| | TMI 1 (B&W) | X | |
| | Calvert Cliffs 1 & 2 (CE) | X | |
| | Ginna (W) | X | |
| FirstEnergy Nuclear Operating Co. | Beaver Valley 1 & 2 (W) | X | |
| | Davis-Besse (B&W) | X | |
| Florida Power & Light \ NextEra | St. Lucie 1 & 2 (CE) | X | |
| | Turkey Point 3 & 4 (W) | X | |
| | Seabrook (W) | X | |



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**PWR Owners Group**
**United States Member Participation* for PA-MSC-1567 (cont.)**

| Utility Member | Plant Site(s) | Participant | |
|---|---|---|---|
| | | Yes | No |
| Florida Power & Light \ NextEra | Pt. Beach 1 & 2 (W) | X | |
| Luminant Power | Comanche Peak 1 & 2 (W) | X | |
| Pacific Gas & Electric | Diablo Canyon 1 & 2 (W) | X | |
| PSEG – Nuclear | Salem 1 & 2 (W) | X | |
| South Carolina Electric & Gas | V.C. Summer (W) | X | |
| So. Texas Project Nuclear Operating Co. | South Texas Project 1 & 2 (W) | X | |
| Southern Nuclear Operating Co. | Farley 1 & 2 (W) | X | |
| | Vogtle 1 & 2 (W) | X | |
| Tennessee Valley Authority | Sequoyah 1 & 2 (W) | X | |
| | Watts Bar 1 & 2 (W) | X | |
| Wolf Creek Nuclear Operating Co. | Wolf Creek (W) | X | |
| Xcel Energy | Prairie Island 1 & 2 (W) | X | |

* **Project participants as of the date the final deliverable was completed. On occasion, additional members will join a project. Please contact the PWR Owners Group Program Management Office to verify participation before sending this document to participants not listed above.**



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**PWR Owners Group**
**International Member Participation* for PA-MSC-1567**

| Utility Member | Plant Site(s) | Participant | |
|---|---|---|---|
| | | Yes | No |
| Asociación Nuclear Ascó-Vandellòs | Asco 1 & 2 (W) | X | |
| | Vandellos 2 (W) | X | |
| Axpo AG | Beznau 1 & 2 (W) | X | |
| Centrales Nucleares Almaraz-Trillo | Almaraz 1 & 2 (W) | X | |
| EDF Energy | Sizewell B (W) | X | |
| Electrabel | Doel 1, 2 & 4 (W) | X | |
| | Tihange 1 & 3 (W) | X | |
| Electricite de France | 58 Units | X | |
| Elektriciteits Produktiemaatschappij Zuid-Nederland | Borssele 1 (Siemens) | X | |
| Eletronuclear-Eletrobras | Angra 1 (W) | X | |
| Emirates Nuclear Energy Corporation | Barakah 1 & 2 | X | |
| Eskom | Koeberg 1 & 2 (W) | X | |
| Hokkaido | Tomari 1, 2 & 3 (MHI) | X | |
| Japan Atomic Power Company | Tsuruga 2 (MHI) | X | |
| Kansai Electric Co., LTD | Mihama 3 (W) | X | |
| | Ohi 1, 2, 3 & 4 (W & MHI) | X | |
| | Takahama 1, 2, 3 & 4 (W & MHI) | X | |
| Korea Hydro & Nuclear Power Corp. | Kori 1, 2, 3 & 4 (W) | X | |
| | Hanbit 1 & 2 (W) | X | |
| | Hanbit 3, 4, 5 & 6 (CE) | X | |
| | Hanul 3, 4 , 5 & 6 (CE) | X | |
| Kyushu | Genkai 2, 3 & 4 (MHI) | X | |
| | Sendai 1 & 2 (MHI) | X | |
| Nuklearna Electrarna KRSKO | Krsko (W) | X | |
| Ringhals AB | Ringhals 2, 3 & 4 (W) | X | |



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| | | | |
|---|---|---|---|
| Shikoku | Ikata 1, 2 & 3 (MHI) | X | |
| Taiwan Power Co. | Maanshan 1 & 2 (W) | X | |

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## LIST OF ACRONYMS AND ABBREVIATIONS

| | |
|---|---|
| ANO-1 | Arkansas Nuclear One Unit 1 |
| ASME | American Society of Mechanical Engineers |
| B&W | Babcock & Wilcox |
| B&WOG | Babcock & Wilcox Owners Group |
| BB | Baffle-to-Baffle |
| BF | Baffle-to-Former |
| BMI | Bottom Mounted Instrumentation |
| BWR | Boiling Water Reactor |
| BWRVIP | Boiling Water Reactor Vessel Internals Program |
| CASS | Cast Austenitic Stainless Steel |
| CE | Combustion Engineering |
| CEA | Control Element Assembly |
| CF | Core Barrel-to-Former |
| CGR | Crack Growth Rate |
| CLB | Current Licensing Basis |
| COD | Crack Opening Displacement |
| CR-3 | Crystal River Unit 3 |
| CRGT | Control Rod Guide Tube |
| CSS | Core Support Shield |
| CW | Cold Worked |
| DB-1 | Davis-Besse Unit 1 |
| DNB | Departure from Nucleate Boiling |
| dpa | displacements per atom |
| EFPY | Effective Full-Power Year |
| EPFM | Elastic-Plastic Fracture Mechanics |
| EPRI | Electric Power Research Institute |
| ET | Electromagnetic Testing (Eddy-Current) |
| FD | Flow Distributor |
| FEA | Finite Element Analysis |
| FEM | Finite Element Model |
| FMEA | Failure Modes and Effects Analysis |
| HAZ | Heat-Affected Zone |
| HWC | Hydrogen Water Chemistry |
| I&E | Inspection and Evaluation |
| IASCC | Irradiation-Assisted Stress Corrosion Cracking |
| IC | Irradiation-enhanced Creep |
| ID | Inside Diameter |
| IE | Irradiation Embrittlement |
| IGSCC | Intergranular Stress Corrosion Cracking |
| IMI | Incore Monitoring Instrumentation |
| IP | Issue Program |
| ISI | In-service Inspection |
| ISR | Irradiation-enhanced Stress Relaxation |
| JCO | Justification for Continued Operation |



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## LIST OF ACRONYMS AND ABBREVIATIONS (cont.)

| | |
|---|---|
| LCB | Lower Core Barrel |
| LEFM | Linear-Elastic Fracture Mechanics |
| LOCA | Loss-of-Coolant Accident |
| LTS | Lower Thermal Shield |
| MRP | Materials Reliability Program |
| MSC | Materials Subcommittee |
| NEI | Nuclear Energy Institute |
| NDE | Non-Destructive Examination |
| NRC | U.S. Nuclear Regulatory Commission |
| NSSS | Nuclear Steam Supply System |
| OD | Outside Diameter |
| OE | Operating Experience |
| ONS | Oconee Nuclear Station |
| PWR | Pressurized Water Reactor |
| PWROG | Pressurized Water Reactor Owners Group |
| PWSCC | Primary Water Stress Corrosion Cracking |
| RICT | Reactor Internals Core Team |
| RIFG | Reactor Internals Focus Group |
| SCC | Stress Corrosion Cracking |
| SCF | Stress Concentration Factor |
| SSE | Safe Shutdown Earthquake |
| SSHT | Surveillance Specimen Holder Tube |
| TAC | Technical Assignment Control |
| TBD | To Be Determined |
| TE | Thermal Embrittlement |
| TLAA | Time-Limited Aging Analysis |
| TMI-1 | Three Mile Island Unit 1 |
| UCB | Upper Core Barrel |
| UT | Ultrasonic Test |
| UTS | Upper Thermal Shield |
| WOG | Westinghouse Owners Group |



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# ACKNOWLEDGEMENTS

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# 1 OBJECTIVE

The objective of this project is to identify consistent, industry-wide analytical methodologies and data requirements for developing:

1. **Acceptance Criteria** for the Primary and Expansion Components identified in the Materials Reliability Program (MRP) Reactor Internals Inspection and Evaluation (I&E) Guidelines (MRP-227)

2. **Evaluation Procedures** for utilities to assess potential safety and functional impacts of degradation in components with observed relevant conditions

These criteria and procedures must be established and generally accepted across the industry. This effort supports the I&E recommendations of MRP-227. The project is to develop acceptance criteria methodology and data requirements. The potential for generic efforts for each component is also evaluated in this program, and where practicable, work scope to address the generic analyses of these components is identified. Because the component lists are specific to the original nuclear steam supply system (NSSS) suppliers, it is anticipated that any effort to develop acceptance criteria for specific components will be submitted to the Pressurized Water Reactor Owners Group (PWROG) for cafeteria funding.

For each of the Primary and Expansion Components listed in MRP-227, this report outlines:

- Type of analyses required
- Required evaluation procedures
- Data required to support analysis
- Flowchart illustrating evaluation path and potential disposition options
- Component items (Primary and Expansion) that can be addressed on a generic basis

The purpose of Revision 3 of this document is to define methodologies and data requirements that are consistent with Revision 1 of MRP-227 [36] as applicable to initial license renewal (60-year operating license). At the time of writing of this document, the MRP-227 inspection and evaluation guidelines are in development for subsequent license renewal (80-year operating license), and are expected to continue to evolve based upon lessons learned from operating experience. Therefore, each utility must remain cognizant of changes made to the methodologies defined herein, and are advised to update the acceptance criteria when substantive changes occur, especially for changes that are mandated by NEI 03-08 [45]. Note that the methodologies defined herein can also be used in conjunction with Rev. 0 of MRP-227-A [1] after accounting for differences in component applicability, component nomenclature, and component numbering.

The intended goal of this program is to develop methodologies that will remain applicable to both initial and subsequent license renewal. Therefore, the methodologies as defined herein are generally either independent of time or consider time dependency when appropriate. Clarification of this intent was made within each methodology by replacing “60-year operating license,” as stated in methodologies in WCAP-17096-NP-A, Rev. 2, with “operating license,” as contained herein. Note that this change is based upon



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the current state of knowledge of the degradation mechanisms, and is subject to changes or additional requirements in future versions of MRP-227.

An additional objective of Revision 3 of this document is to replace the content repeated in WCAP-17096 from MRP-227 with references to MRP-227. This change provides simplification, clarifies which requirements are defined within MRP-227, and avoids inadvertent application of out-of-date information.

## 1.1 NEI 03-08 ELEMENTS

The management of materials issues is governed by NEI 03-08 [45], which outlines the policies and practices that the industry commits to follow in managing materials aging issues. The industry supports the utilities through participation in Issue Programs. An important output from the Issue Program is the recommended classification of each material issue as either Mandatory, Needed, or Good Practice, which is dependent on the relative level of importance of the issue and associated recommended action.

WCAP-17096 is a work product of the PWROG Materials Subcommittee (MSC) Issue Program. It is intended to be a companion document to MRP-227, which is a work product of the EPRI Materials Reliability Program (MRP) Issue Program, and contains the 'Needed' requirement for utilities to implement MRP-227 Table 4-1 through Table 4-9 and Table 5-1 through Table 5-3 for the applicable design into the plant's reactor internals aging management program. WCAP-17096 provides the acceptance criteria methodology for addressing relevant indications observed during MRP-227 inspections of reactor internals components. There are no specific NEI 03-08 Mandatory, Needed, or Good Practice elements in this document. However, the methodologies defined herein can be used to meet the following NEI 03-08 Needed requirement in Section 7.5 of MRP-227 [1]:

> "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, WCAP-17096-NP or equivalent methods)."

It is noted that an alternate methodology may be acceptable with adequate technical justification.



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# 2 BACKGROUND

The MRP Reactor Internals Focus Group (RIFG) issued a formal request for the PWROG to sponsor a project to develop "generic" acceptance criteria for the proposed MRP-227 I&E Guidelines [1] on March 19, 2008 (Reference MRP Letter 2008-026). The inspection and evaluation recommendations in MRP-227 were issued under the Nuclear Energy Institute (NEI) NEI-03-08 protocol for industry wide implementation. MRP-227 was also submitted to the U.S. Nuclear Regulatory Commission (NRC) with a request for a full safety evaluation (SE) report. The NRC completed a Revision 1 for the SE on MRP-227, Revision 0. The SE is incorporated into MRP-227-A [1]. In their review of WCAP-17096, the NRC also requested [15] that the Electric Power Research Institute (EPRI) revise the WCAP-17096 report in accordance with the SE, which was completed in WCAP-17096-NP-A, Rev. 2.

The MRP-227 I&E Guidelines in [1] for pressurized water reactor (PWR) internals are based on a thorough screening of both potential degradation and operating experience [2, 3, and 4]. They are designed to target inspections of locations where aging degradation can potentially impair component function. Engineering evaluations and assessments (also known as functionality analyses) associated with the original screening evaluations have identified the potential operational concerns, and inspection methodologies have been identified for each component [5, 6, and 49]. However, the current I&E Guidelines do not provide detailed acceptance and evaluation criteria for each component.

The I&E Guidelines in MRP-227 build on the existing ASME Code, Section XI inspections to create comprehensive inspection recommendations for aging degradation in reactor internals [7, 8]. The fundamental goal of any inspection under the ASME Code, Section XI program [9] is to identify relevant conditions that require further action. Article IWA-9000 of the ASME Code defines relevant condition as follows:

> Relevant Condition – A condition observed during a visual examination that requires supplemental examination, corrective measure, correction by repair/replacement activities, or analytical evaluation.[1]

The ASME Code [9] defines acceptance standards in IWB-3400, which are used to determine whether an observed condition is acceptable for service or is a relevant condition. The inspection standards used to define the relevant condition are based on generic analysis that provides a high level of assurance of satisfactory function. Acceptance standards for ASME Examination Categories B-N-2 (Welded Core Support Structures and Interior Attachments to Reactor Vessels) and B-N-3 (Removable Core Support Structures) are provided in IWB-3520. While these acceptance standards are appropriate for the ASME Code Section XI inspections, including those that are specifically highlighted in MRP-227 as existing reactor internals inspection requirements, they are not fully applicable to the MRP-227 inspection recommendations. In particular, the linear flaw standards of IWB-3510, as intended to guard against propagation of cracks through the reactor pressure vessel, are not meaningful when applied to the removable internals components that do not provide a pressure boundary.

[1] ASME (Article IWA-9000) uses this definition for visual examinations. The definition has been expanded here to include all inspections conducted under the I&E Guidelines.



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MRP-227 provides lists of components with specific inspection recommendations. The components are divided into three basic categories based on the recommendation:

- Primary Components – Inspection of the Primary Components is required in the I&E Guidelines. In general, these inspections must be conducted early in the license extension period. Every plant needs to have acceptance criteria for Primary Components included in their aging management program.

- Expansion Components – MRP-227 defines criteria for the results of Primary Component inspections that will trigger inspection of the Expansion Components. Although the MRP-227 requirements allow a time delay for mobilization of equipment and resources, acceptance criteria for the expansion components will also be required.

- Existing Components – There are established acceptance criteria for the ASME Code, Section XI and other exams in this category. The PWROG should continue to review the current field practice on the ASME Code Section XI inspections as part of the implementation evaluations to ensure a consistent approach where industry guidance has not already addressed ambiguity in IWB-2500.

Each of these components also requires some acceptance standard for evaluating the relevant condition. For many general condition monitoring visual examinations (VT-3), MRP-227 identifies specific conditions that supplement the list of relevant conditions provided in IWB-3520.2. However, there is not a comprehensive listing of acceptance standards in MRP-227. The general standards for determining relevant conditions in reactor internals inspections are provided in MRP-228 [10]:

- Welds – Cracks, or indications that exhibit characteristics of cracking, are considered relevant.

- Components – Cracking or other significant degradation that could impair the ability of the component to perform its design function is considered relevant.

Once a relevant condition has been identified, evaluation is required to assess the ability of the degraded component to continue to perform the design function without interfering with the function of the system. The evaluation of suitability for continued service will be, in general, an extension of the analysis used to define a relevant condition. The scope of this PWROG Reactor Internals Core Team (RICT) Project was to define the process for developing the acceptance criteria and evaluation procedures. The process is defined in the following terms:

- Evaluation Methodology – The procedures and criteria to be used by the engineering staff to evaluate relevant conditions. This includes:

  - Demonstration of functionality of the current configuration
  - Establishment of a re-inspection frequency of one or more refueling cycles
  - An engineering basis for repair/replacement/mitigation options

Note: Detailed repair/replacement/mitigation options are outside the scope of this document.



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- Acceptance Criteria – The criteria against which the need for corrective action will be evaluated. The acceptance criteria should ensure that the intended functions of the particular structure and component are maintained under all current licensing basis design conditions during the period of extended operation.[2]

In some cases, it will be feasible to avoid plant-specific evaluations by adopting generic standards for acceptance that will ensure compliance with the accepted industry requirements.

- Generic Acceptance – Disposition of a relevant condition based on a generic implementation of the evaluation methodology for an NSSS design or other plant grouping.

The original MRP request forwarded to the PWROG RICT was for generic acceptance criteria. However, as a prerequisite to developing the generic criteria, the evaluation methodology must be defined. The development of evaluation methodologies also poses significant implementation issues that need to be considered by the PWROG. This report outlines the evaluation methodology for components to be inspected under the I&E Guidelines and identifies analyses that may be conducted on a generic basis. Separate PWROG projects support the development of evaluation methodologies and generic acceptance criteria.

## 2.1 NRC REVIEW OF WCAP-17096

Letter OG-09-290 [33] issued Revision 0 of this document for review and comment by the PWROG Materials Subcommittee on July 27, 2009. Comments from this review were incorporated in Revision 1 and Revision 2 of the document.

In May 2010, EPRI transmitted WCAP-17096, Revision 2 to the USNRC for safety evaluation in [15]. The report was accepted by the NRC for detailed review as documented in [17].

The NRC issued requests for additional information (RAIs) on the document in [18], and updated in [19]. These RAIs were responded to by EPRI via [20], with revised responses transmitted in [21].

The NRC issued additional RAIs via [22], which were responded to by EPRI with transmittal of [23], and responses were updated and transmitted via [24].

As a result of the RAIs issued, in addition to lessons learned from plant-specific work performed by Framatome and Westinghouse, required changes were identified to the version of WCAP-17096-NP that was submitted for review. A significant portion of these changes were to the appendices of the report that were updated and included as attachments to [25, 26, 27, and 28]. These documents were submitted to the NRC as a means to identify specific changes that would be made to the final version of the report.

Following submittal of these letters, the NRC issued two additional RAIs via [29] which were responded to by EPRI in [30]. Upon receipt of the responses the NRC issued four additional RAIs via [31]. EPRI responded to these RAIs in [32].

The NRC issued its final safety evaluation on WCAP-17096-NP, Revision 2 in May 2016 [34].

[2] Definition based on [12] Aging Management Program (AMP), Element 6.



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The approved version of this topical report, WCAP-17096-A, Revision 2, was transmitted to the NRC in September 2016 via OG-16-299 [35]. The approved version includes all changes that were identified and documented throughout the NRC review of the document. Specifically, all document markups that were transmitted to the NRC in response to RAIs were incorporated into the approved version of the topical report.

## 2.2 ADDRESSING CONDITIONS FROM THE SAFETY EVALUATION IN WCAP-17096-NP-A, REV. 2

A safety evaluation (SE) was issued in [34] based upon the review of WCAP-17096-NP, Rev. 2 by the NRC staff. The final approved version of the topical report, as transmitted in OG-16-299 [35] included the SE, the topical report itself, and the correspondence between the NRC staff and the industry during the development of the approved version. Section 4.0 of the SE identifies several conditions imposed by the NRC staff on utility implementation of WCAP-17096-NP, Rev. 2. The following section explains how each condition defined in the SE [34] to WCAP-17096-NP, Rev. 2 is addressed in WCAP-17096-NP, Rev. 3.

Condition 1 (B&W RVI Components)

> *"Condition 1 regards the schedule for submitting to the NRC, analyses, justification for operation, replacement schedule, etc. to be performed by EPRI or licensees to support the inspection specified in the TR for various RVI component items. For easy referencing, the NRC staff places the RVI component items into three groups under Condition 1:*
>
> *For Group 1 RVI component items*
>
> *As discussed in Section 3.2.1 of this SE, the licensee is required to submit the detailed analysis, replacement schedule, or justification for some other alternative process, for the three inaccessible Expansion component items within one year of the inspection of the linked Primary component items for NRC to determine whether review and approval are needed if the inspection results indicate aging triggering the expansion criteria in Table 5-1 of MRP-227-A.*
> *The three inaccessible Expansion component items are:*
>
> - *the core barrel cylinder and welds,*
> - *the former plates, and*
> - *the CF bolts, the internal and external BB bolts, and the locking devices for CF and external BB bolts.*
>
> *For future applications implementing MRP-227-A, the plant-specific inspection plan can use Condition 1/Group 1 in this SE, instead of the more restrictive Applicant/Licensee Action Item 6 in the MRP-227-A SE. Therefore, in this case, Condition 1/Group 1 is a replacement of Applicant/Licensee Action Item 6. Additionally, as discussed under "Core Barrel Assembly Plates" in Section 3.2.2.1 of this SE, the licensee is required to submit the LEFM analysis for the baffle plates under normal/upset condition loads, considering IASCC, within one year of the inspection of the Primary component item for NRC to determine whether review and approval are needed if the inspection results indicate aging not meeting the acceptance criteria in Table 5-1 of MRP-227-A.*



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*For Group 2 RVI component items*

*As discussed in Section 3.2.2.1 of this SE, the licensee is required to submit the following within one year of the inspection of the Primary or linked Primary component items for NRC to determine whether review and approval are needed if the inspection results for the Primary RVI components mentioned in the first two bullets indicate aging not meeting the acceptance criteria in Table 5-1 of MRP-227-A, and for Expansion RVI components mentioned in the next three bullets, if inspection results for their linked Primary RVI components indicate aging triggering the expansion criteria in Table 5-1 of the MRP-227-A:*

- *analysis for the top and bottom retaining rings of the vent valve,*
- *thermal and structural analyses for the FD bolts,*
- *thermal and structural analyses for the UTS and LTS bolts, with the analyses for UTS bolts including evaluation of joint stability of the thermal shield considering the FIV,*
- *the analyses for SSHT studs/nuts or bolts, and*
- *the analysis for TMI-1 lower grid shock pad bolts.*

*For Group 3 RVI component items*

*As discussed under "CRGT Spacer Castings" and "IMI Guide Tube Spiders and Welds" in Section 3.2.2.1 of this SE, the licensee is required to submit the plant-specific functionality analyses for B&W IMI guide tube assembly spiders and CRGT spacer castings within one year[1] of the inspection of these components for NRC to determine whether review and approval are needed if the inspection results indicate aging not meeting the acceptance criteria in Table 5-1of MRP-227-A."*

Condition 1 Resolution

Group 1 RVI component items are addressed in WCAP-17096, Rev. 3 by the addition of the following statement in Appendix A, Sections A.2.5, A.2.6, A.2.7, A.2.8, and A.2.9: "The licensee shall submit the plant-specific analysis, replacement schedule, or justification for some other alternative process for this component item within one year after the inspection of the linked Primary item when the results trigger the expansion criteria in MRP-227."

Group 1 RVI component items are also addressed in WCAP-17096, Rev. 3 by the addition of the following statement in Appendix A, Section A.1.9: "The licensee is required to submit the LEFM analysis for the baffle plates under normal/upset condition loads, considering IASCC, within one year of the inspection of the Primary component items for NRC to determine whether review and approval are needed if the inspection results indicate aging not meeting the acceptance criteria in Table 5-1 of MRP-227."

Group 2 RVI Primary component items are addressed in WCAP-17096, Rev. 3 by the addition of the following statement in Appendix A, Sections A.1.3, A.1.11, and A.1.13: "The licensee shall submit the plant-specific analysis for this component item within one year after the inspection if the results do not meet the acceptance criteria in MRP-227."

Group 2 RVI Expansion component items are addressed in WCAP-17096, Rev. 3 by the addition of the following statement in Appendix A, Sections A.2.3, A.2.4, and A.2.13: "The licensee shall submit the



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plant-specific analysis for this component item within one year after the inspection of the linked Primary item when the results trigger the expansion criteria in MRP-227."

Group 3 RVI component items are addressed in WCAP-17096, Rev. 3 by the addition of the following statement in Appendix A, Sections A.1.2 and A.1.14: "The licensee shall submit the plant-specific analysis for this component item within one year after the inspection if the results do not meet the acceptance criteria in MRP-227."

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

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

Condition 2 Resolution

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

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

This statement has been added to the following components:

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



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- CE-ID: 6.1, Core Support Barrel Assembly – Middle Axial Weld
- CE-ID: 6.2, Core Support Barrel Assembly – Lower Axial Weld
- W-ID: 3.1, Core Barrel Assembly – Upper Girth Weld
- W-ID: 3.2, Core Barrel Assembly – Upper Axial Weld
- W-ID: 4, Core Barrel Assembly – Lower Girth Weld
- W-ID: 4.2, Core Barrel Assembly – Middle Axial Welds
- W-ID: 4.3, Core Barrel Assembly – Lower Axial Welds

Therefore, Condition 2 has been addressed within the methodologies as defined in WCAP-17096-NP, Rev. 3. It is noted that MRP-227, Rev. 1 revised the nomenclature for several Primary and Expansion components and modified the Primary to Expansion linkages for some components. Therefore, the naming and identification numbers from the WCAP-17096-NP-A, Rev. 2 SE conditions may not exactly match those in the current revision of the topical report.

Note that the CE component, "Core Support Barrel Assembly (Welded) – Ribs and Rings," has been identified as an inaccessible component for MRP-227, Rev. 1 as discussed in Part C of RAI 12 [41]. This change to MRP-227, Rev. 1 was accepted by the NRC in the final SE for MRP-227, Rev. 1 [57]. Since a fracture mechanics analysis is no longer applicable to this component, the Condition to define the high fluence weld CGR is no longer applicable to this component.

Condition 3 (CE and Westinghouse RVI Components)

> "*Condition 3 regards the schedule for submitting to the NRC the analyses, justification for operation, replacement schedule, etc. to be performed by EPRI or licensees to support the inspection specified in the TR for various RVI component items.*
>
> *As discussed under Section 3.2.2.2.2 of this SE, the applicant is required to submit the analyses described for each group below for the NRC to determine if review and approval is needed, within one year[1] after (1) any inspection for which the results trigger the expansion criteria of MRP-227-A Tables 5-2 or 5-3 (for components that have expansion criteria) or; (2) any inspection that detects relevant conditions as defined in MRP-227-A, Tables 5-2 or 5-3 (for components that do not have expansion criteria).*
>
> *The one year time period shall not be used as justification for delaying submittals which are required to support continued operation under the current regulations.*
>
> *For Group 1 RVI Component Items:*
>
> *The licensee shall submit the plant-specific or generic FMEA analysis and confirmation of applicability of generic FMEA analyses, if a generic FMEA is referenced, within one year[1] after any inspection that detects relevant conditions as defined in MRP-227-A, Tables 5-2 or 5-3, for:*
>
> *• W-ID: 6 Baffle-Former Assembly - Baffle edge bolts*
> *• CE-ID: 4 Core Shroud Assembly (Bolted) - Assembly*
> *• W-ID: 8 Baffle-Former Assembly - Assembly*
> *• CE-ID: 5 Core Shroud Assembly (Welded) – Assembly*
>
> *For Group 2 RVI Component Items:*



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*The licensee shall submit the plant-specific analysis for the acceptable minimum distribution of intact components within one year[1] after (1) any inspection for which the results trigger the expansion criteria of MRP-227-A Tables 5-2 or 5-3 (for components that have expansion criteria) or; (2) any inspection that detects relevant conditions as defined in MRP-227-A, Tables 5-2 or 5-3 (for components that do not have expansion criteria).*

*The components with expansion criteria covered by Condition 3, Group 2 are:*

- *CE-ID: 1.2 Core Shroud Assembly (Bolted) - Core Support Column Bolts*
- *W-ID: 7.2 – Lower Support Assembly – Lower Support Column Bolts*

*The components that do not have expansion criteria covered by Condition 3, Group 2 are:*

- *CE-ID: 8 Lower Support Structure – Core Support Column Welds*
- *W-ID: 2.3 – Lower Support Assembly – Lower Support Column Bodies (Cast)[2],*
- *W-ID: 3.2 – Lower Support Assembly – Lower Support Columns (Non- Cast)*

*For Group 3 RVI Component Items listed below:*

*a) In order to apply the TR methodology for cracks detected via one-sided visual examinations, if supplementary examinations to determine crack depth were not performed, a follow up examination must be performed no later than the next RFO to confirm the assumed crack growth rates are conservative, unless a technical justification for a longer verification interval is submitted.*

*b) The licensee shall submit the plant-specific crack growth and fracture mechanics analysis justifying the detected flaws, including a justification of the allowable operating period before re-inspection, within one year[1] after (1) any inspection for which the results trigger the expansion criteria of MRP-227-A Tables 5-2 or 5-3 (for those components with expansion criteria) or; (2) any inspection that detects relevant conditions as defined in MRP-227-A, Tables 5-2 or 5-3 (for those components that do not have expansion criteria).*

*The components that have expansion criteria covered by Condition 3, Group 3, Item b are:*

- *CE-ID: 6 Core Support Barrel Assembly – Upper (Core Support Barrel) Flange Weld*
- *CE-ID: 7 Core Support Barrel Assembly – Lower Cylinder Girth Welds*
- *W-ID: 3 Core Barrel Assembly – Upper Core Barrel Flange Weld*
- *W-ID: 3.1 Core Barrel Assembly – Core Barrel Outlet Nozzle Welds*
- *W-ID: 4 Core Barrel Assembly – Upper and Lower Core Barrel Cylinder Girth Welds*

*The components that do not have expansion criteria covered by Condition 3, Group 3, Item b are:*

- *CE-ID: 6.1 Core Support Barrel Assembly – Lower Core Barrel Flange*
- *CE-ID: 6.2 Core Support Barrel Assembly – Upper Cylinder (Including Welds)*
- *CE-ID: 7.1 Core Support Barrel Assembly – Core Barrel Assembly Axial Welds*
- *CE-ID: 9 Core Support Barrel Assembly – (Lower Flange Flexure Weld)*
- *W-ID: 4.1 Core Barrel Assembly – Upper and Lower Core Barrel Cylinder Axial Welds*
- *W-ID: 5 Core Barrel Assembly – Lower Core Barrel Flange Weld*
- *CE-ID: 6.3 Core Support Barrel Assembly – Upper Core Barrel Flange*
- *CE-ID: 6.4 Lower Support Structure – Lower Core Support Beams*



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*For Group 4 RVI Components*

*The licensee shall submit the plant-specific fracture mechanics analysis justifying the detected flaws within one year[1] after (1) any inspection for which the results trigger the expansion criteria of MRP-227-A Tables 5-2 or 5-3 (for those components with expansion criteria) or; (2) any inspection that detects relevant conditions as defined in MRP-227-A, Tables 5-2 or 5-3 (for those components that do not have expansion criteria).*

*The components with expansion criteria covered by Condition 3, Group 4 are:*
*• CE-ID: 2 - Core Shroud Assembly (Welded) - Core Shroud Plate-Former Plate Weld*
*• CE-ID: 3 Core Shroud Assembly (Welded - Full Height) - Shroud Plates*

*The components without expansion criteria covered by Condition 3, Group 4 are:*

*• CE-ID: 2.1- Core Shroud Assembly (Welded) - Remaining Axial Welds*
*• CE-ID: 3.1 Core Shroud Assembly (Welded) – Remaining Axial Welds, Ribs and Rings*

*For Group 5 RVI Components*

*• For W-ID: 2 Control Rod Guide Tube Assembly, the plant-specific analysis of acceptable intact weld patterns for the CRGT lower flange welds be submitted for the NRC to determine if review and approval is needed, within one year[1] after any inspection for which the results trigger the expansion criteria of MRP-227-A, Table 5-3.*

*• For W-ID: 10 Thermal Shield Assembly - Thermal Shield Flexures, the plant-specific analysis of acceptable dynamic response of the remaining flexures be submitted for the NRC to determine if review and approval is needed, within one year[1] after any inspection that detects relevant conditions as defined in the "Examination Acceptance Criteria" column of MRP-227-A, Table 5-3.*

*For Group 6 items:*

*The licensee shall submit the plant-specific analysis for the following RVI component items within one year[1] after any inspection that detects relevant conditions as defined in MRP-227-A, Table 5-2:*
*• W-ID: 2.1 Upper Internals Assembly – Upper Core Plate*
*• W-ID: 2.2 Lower Internals Assembly - Lower Support Forging or Casting*
*• CE-ID: 10 Lower Support Structure – Core Support Plate*
*• CE-ID: 11 Upper Internals Assembly – Fuel Alignment Plate"*

Note that the footnotes cited above are defined as follows:

*"1. The one year period may be considered to begin on the date the plant begins power operation upon startup from the refueling outage during which the degradation was discovered.*
*2. If plant-specific functionality analyses have been performed to respond to applicant/licensee Action Item 7 from MRP-227-A prior to the initial inspection of the components, licensees may compare the inspection findings to the previously performed functionality analyses and need to resubmit plant-specific analyses unless the inspection findings are not bounded by the original analysis."*



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Condition 3 Resolution

Condition 3 has been addressed in the current revision of this topical report by adding requirements statements to the applicable methodologies in Appendices C and E. Condition 3 is related to: 1) requirement to submit the evaluations within one-year when used to assess degradation or trigger expansion criteria, and 2) one-sided visual examination re-inspection requirements. It is noted that MRP-227, Rev. 1 revised the nomenclature for several Primary and Expansion components and modified the Primary to Expansion linkages for some components. Therefore, the naming and identification numbers from the WCAP-17096-NP-A, Rev. 2 SE conditions may not exactly match those in the current revision of the topical report.

Submittal of Evaluations within One-Year

The conditions as defined in the SE to WCAP-17096-NP-A, Rev. 2 stipulate that the analyses used to support continued operation with degradation must be submitted to the NRC within one year after degradation is detected. The components were divided into six groups, based upon the type of analysis. For Primary components with an Expansion link, the submittal requirement is prompted by degradation triggering the expansion criteria. For components without an Expansion Link, the submittal requirement is prompted by the inspection detecting relevant conditions according to MRP-227.

*Group 1 RVI Components*

The following statement has been added to Group 1 RVI components where Table 5-2 applies to CE components and Table 5-3 applies to Westinghouse components:

> *Note that the licensee shall submit the plant-specific or generic FMEA analysis and confirmation of applicability of generic FMEA analysis within one year after any inspection that detects relevant conditions as defined in Table 5-2 (or Table 5-3) of MRP-227.*

This statement has been added to the following components:

- CE-ID: 4, Core Shroud Assembly (Bolted) – Assembly
- CE-ID: 4a, Core Shroud Assembly (Welded) – Assembly
- W-ID: 5, Baffle-Former Assembly – Baffle-Edge Bolts
- W-ID: 7, Baffle-Former Assembly – Assembly

*Group 2 RVI Components*

The following statement has been added to Group 2 RVI components with expansion criteria where Table 5-2 applies to CE components and Table 5-3 applies to Westinghouse components:

> *Note that the licensee shall submit the plant-specific analysis for the acceptable distribution of intact components within one year after any inspection that triggers the expansion criteria as defined in Table 5-2 (or Table 5-3) of MRP-227.*

This statement has been added to the following components:



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- CE-ID: 1.1, Core Shroud Assembly (Bolted) – Core Support Column Bolts
- W-ID: 6.2, Lower Support Assembly – Lower Support Column Bolts

A similar one-year submittal requirement has been added to Group 2 components without expansion criteria where Table 5-2 applies to CE components and Table 5-3 applies to Westinghouse components:

> *Note that the licensee shall submit the plant-specific analysis for the acceptable distribution of intact components within one year after any inspection that detects the relevant conditions as defined in Table 5-2 (or Table 5-3) of MRP-227.*

This statement has been added to the following components:

- CE-ID: 6.3, Lower Support Structure – Core Support Columns
- W-ID: 4.4, Lower Support Assembly – Lower Support Column Bodies (Both Cast and Non-Cast)

Note that "acceptable distribution" of intact components is included rather than "acceptable minimum distribution," since acceptable patterns are determined, but the minimum pattern is not necessarily determined when applying the methodology.

Also note that footnote 2 to the SE to WCAP-17096-NP-A provided the following exception to the submittal requirement for Cast Lower Support Column Bodies (W-ID: 4.4):

> *"If plant-specific functional analyses have been performed to respond to applicant/licensee Action Item 7 from MRP-227-A prior to the initial inspection of the components, licensees may compare the previously performed functionality analyses and need not resubmit plant-specific analyses unless the inspection findings are not bounded by the original analysis."*

A functionality analysis was performed by the PWROG, which is applicable to CE Core Support Columns and Westinghouse Lower Support Column Bodies (Both Cast and Non-Cast). This evaluation fulfilled its intended purpose, and resulted in the closure of Application / Licensee Action Item 7 (Plant-Specific Evaluation of Cast Austenitic Stainless Steels Materials) per [53]. However, the functionality analysis is missing critical elements of the acceptance criteria methodology in WCAP-17096-NP-A, and therefore, cannot be generically applied to disposition inspection findings.

*Group 3 RVI Components*

The submittal requirement for Group 3 RVI Item (b) includes a similar statement as the other RVI groups regarding one-year submittal. An additional requirement is defined for Group 3 RVI to determine the allowable operating period before re-inspection must occur.

Therefore, the following submittal requirement was added to Group 3 RVI components where Table 5-2 applies to CE components and Table 5-3 applies to Westinghouse components:

> *Note that the licensee shall submit the plant-specific crack growth and fracture mechanics analysis justifying the detected flaws, including a justification of the allowable operating period before re-inspection, within one year after any inspection that triggers the expansion criteria as defined in Table 5-2 (or Table 5-3) of MRP-227.*



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This statement has been added to the following components:

- CE-ID: 5, Core Support Barrel Assembly – Upper Flange Weld
- CE-ID: 6, Core Support Barrel Assembly – Middle Girth Weld
- W-ID: 3, Core Barrel Assembly – Upper Flange Weld
- W-ID: 4, Core Barrel Assembly – Lower Girth Weld

Note that the Westinghouse component, "Core Barrel Assembly – Core Barrel Outlet Nozzle Weld," is no longer an in-scope component in MRP-227, Rev. 1. Therefore, the submittal of the evaluations is no longer applicable.

A similar submittal requirement was added to Group 3 RVI Item (b) components without expansion criteria where Table 5-2 applies to CE components and Table 5-3 applies to Westinghouse components:

> *Note that the licensee shall submit the plant-specific crack growth and fracture mechanics analysis justifying the detected flaws, including a justification of the allowable operating period before re-inspection, within one year after any inspection that detects relevant conditions as defined in Table 5-2 (or Table 5-3) of MRP-227.*

This statement has been added to the following components:

- CE-ID: 5.1, Core Support Barrel Assembly – Lower Girth Weld
- CE-ID: 5.2, Core Support Barrel Assembly – Upper Girth Weld
- CE-ID: 5.3, Core Support Barrel Assembly – Upper Axial Weld
- CE-ID: 5.4, Lower Support Structure – Lower Core Support Beams
- CE-ID: 6.1, Core Support Barrel Assembly – Middle Axial Weld
- CE-ID: 6.2, Core Support Barrel Assembly – Lower Axial Weld
- CE-ID: 7, Core Support Barrel Assembly – CSB Flexure Weld
- W-ID: 3.1, Core Barrel Assembly - Upper Girth Weld
- W-ID: 3.2, Core Barrel Assembly – Upper Axial Weld
- W-ID: 3.3, Core Barrel Assembly – Lower Flange Weld
- W-ID: 4.2, Core Barrel Assembly – Middle Axial Welds
- W-ID: 4.3, Core Barrel Assembly – Lower Axial Welds

Note that the CE component, "Core Support Barrel Assembly – Upper Core Barrel Flange," is no longer an in-scope component in MRP-227, Rev. 1. Therefore, the submittal of the evaluations is no longer applicable. Also note that the Westinghouse component, "Core Support Assembly – Upper Girth Weld," has been moved from a Primary to an Expansion component in MRP-227, Rev. 1. Therefore, "detection of relevant conditions," rather than "triggers the expansion criteria" is applicable to this component.

*Group 4 RVI Components*

The following statement has been added to Group 4 RVI components with expansion criteria:



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

This statement has been added to the following components:

- CE-ID: 2, Core Shroud Assembly (Welded) - Core Shroud Plate-Former Plate Weld
- CE-ID: 3, Core Shroud Assembly (Welded) – Shroud Plates

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

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

This statement has been added to the following components:

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

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

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

This statement has been added to the following component:

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

*Group 5 RVI Components*

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

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

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



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

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

*Group 6 RVI Components*

The following statement has been added to Group 6 RVI components without expansion criteria where Table 5-2 applies to CE components and Table 5-3 applies to Westinghouse components:

> *Note that the licensee shall submit the plant-specific analysis for this component within one year after any inspection that detects relevant conditions as defined in Table 5-2 (or Table 5-3) of MRP-227.*

This statement has been added to the following components:

- CE-ID: 9, Lower Support Structure – Core Support Plate
- CE-ID: 10, Upper Internals Assembly – Fuel Alignment Plate
- W-ID: 3.4, Lower Internals Assembly – Lower Support Forging or Casting
- W-ID: 4.1, Upper Internals Assembly – Upper Core Plate

One-Sided Visual Examination Re-inspection Requirement

The condition for Item (a) of Group 3 RVI in the SE to WCAP-17096-NP-A for the core barrel weld acceptance criteria defines a limitation of the EVT-1 inspection method as used during the one-sided visual examinations of the core barrel welds. Re-inspection must occur at the next refueling outage unless supplemental inspections are performed to determine the crack depth or justification for a longer verification interval is submitted. Note that confirmation of the predicted crack growth rate is a separate requirement as defined within each evaluation methodology which is not to be excluded solely on determination of the crack depth.

Therefore, the following statement has been added to the methodologies for the applicable components:

> *In order to apply the TR methodology for cracks detected via one-sided visual examinations, if supplementary examinations to determine crack depth were not performed, a follow up examination must be performed no later than the next RFO to confirm the assumed crack growth rates are conservative, unless a technical justification for a longer verification interval is submitted.*

This statement has been added to the following Group 3 components, which apply crack growth and fracture mechanics analysis:

- CE-ID: 5, Core Support Barrel Assembly – Upper Flange Weld
- CE-ID: 5.1, Core Support Barrel Assembly – Lower Girth Weld



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- CE-ID: 5.2, Core Support Barrel Assembly – Upper Girth Weld
- CE-ID: 5.3, Core Support Barrel Assembly – Upper Axial Weld
- CE-ID: 5.4, Lower Support Structure – Lower Core Support Beams
- CE-ID: 6, Core Support Barrel Assembly – Middle Girth Weld
- CE-ID: 6.1, Core Support Barrel Assembly – Middle Axial Weld
- CE-ID: 6.2, Core Support Barrel Assembly – Lower Axial Weld
- CE-ID: 7, Core Support Barrel Assembly – CSB Flexure Weld
- W-ID: 3, Core Barrel Assembly – Upper Flange Weld
- W-ID: 3.1, Core Barrel Assembly - Upper Girth Weld
- W-ID: 3.2, Core Barrel Assembly – Upper Axial Weld
- W-ID: 3.3, Core Barrel Assembly – Lower Flange Weld
- W-ID: 4, Core Barrel Assembly –Lower Girth Weld
- W-ID: 4.2, Core Barrel Assembly – Middle Axial Welds
- W-ID: 4.3, Core Barrel Assembly – Lower Axial Welds

Note that the CE component, "Core Support Barrel Assembly – Upper Core Barrel Flange," and Westinghouse component, "Core Barrel Assembly – Core Barrel Outlet Nozzle Weld," are no longer in-scope components in MRP-227, Rev. 1. Therefore, the restriction on one-sided inspections is not applicable since these components are not inspected.

## 2.3 INDUSTRY ACTIVITY SINCE NRC APPROVAL OF WCAP-17096-NP, REV. 2

Since the approved version of WCAP-17096-NP was submitted on the docket, the companion document, MRP-227, has been updated to MRP-227, Revision 1 [36]. MRP-227, Revision 1 was submitted to the NRC in December of 2015 [37] and was accepted by the NRC for safety evaluation review [38] in July 2016. The NRC issued RAIs [39] in May 2017, which EPRI responded to via MRP 2017-027 [40] in October 2017 and 2018-003 [41] in January 2018. Additional RAIs on recent plant operating experience were provided by the NRC staff in July 2018 [55], which were responded to in [56]. Changes associated with MRP-227, Revision 1 were included in the present document based on NRC concurrence with the MRP-227, Rev. 1 changes as accepted in the Final Safety Evaluation for MRP-227, Rev. 1 [57].

Following the observation of indications on a large number of baffle-former bolts at multiple Westinghouse-designed plants, the industry developed interim guidance on MRP-227 to address the issue. NEI 03-08 Needed guidance for baffle bolt inspections was issued by the EPRI Materials Reliability Program (MRP) via MRP 2016-021 [42] and MRP 2017-009 [43]. The NRC issued questions on the Interim Guidance, which were responded to via [44]. A staff assessment of the baffle-former bolt interim guidance was completed in [54].

As a first step towards the current update of the acceptance criteria methodologies, interim guidance was compiled into PWROG-17071-NP [59] for methodologies requiring changes, which was transmitted to the NRC for information in OG-18-168 [58]. The changes contained in PWROG-17071 included: additional methodologies for B&W components, realignment of component categorization for CE and Westinghouse plants due to primary and expansion component categorization changes in MRP-227, Revision 1, changes to the Westinghouse baffle-former evaluation methodologies, as well as additional miscellaneous changes.



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Section 2.3.2 of PWROG-17071-NP contained the following submittal requirement for Westinghouse baffle-former bolts:

> 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.

As discussed in RAI 30 in MRP 2018-026 [56], the submittal requirement in PWROG-17071 did not meet the full intent of the NRC Staff Assessment in [54]. The updated submittal guidance has been included as applicant licensee action item (A/LAI) 1 in the SE to MRP-227, Rev. 1 [57]. Therefore, the submittal requirement for baffle-former bolts for Westinghouse plants (W-ID: 6) was updated to 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 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

Interim guidance for the guide card wear methodology was issued in OG-18-46 [60] as applicable to the methodology in WCAP-17451-P, Rev. 1 [61]. The interim guidance has been incorporated into the updated acceptance criteria methodology contained herein, referencing WCAP-17451-P, Rev. 2 [47].



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## 2.4 CHANGE SUMMARY FOR WCAP-17096, REV. 3

The purpose of this section is to identify the changes introduced into WCAP-17096-NP, Rev. 3 relative to the initial NRC-approved version (WCAP-17096-NP-A, Rev. 2) that was issued in August 2016. Table 2-1 identifies the changes made in WCAP-17096-NP, Rev. 3 as issued in July 2019 for submittal to the NRC for review and approval. Table 2-2 identifies the changes made in WCAP-17096-NP-A, Rev. 3 due to the incorporation of RAI responses and the updates due to NRC-approval of Code Case N-889 [52] with Conditions.

**Table 2-1 Summary of Changes in WCAP-17096-NP, Rev. 3 relative to WCAP-17096-NP-A, Rev. 2**

| Section | Description of Change |
|---|---|
| All | • Changed from "AREVA" to "Framatome"<br>• Referred to MRP-227 for appropriate content<br>• Editorial changes and clarifying text where appropriate |
| 1 | • Added purpose of Rev. 3 of document<br>• Clarified intended goal of methodology applicability<br>• Additional objective is to remove content repeated from MRP-227 and replace with reference to MRP-227 |
| 1.1 | Explained how NEI 03-08 elements are addressed |
| 2 | Provided clarification on existing inspections |
| 2.2 | Added a discussion of how each Condition in the Safety Evaluation for WCAP-17096-NP-A, Rev. 2 was addressed in WCAP-17096-NP, Rev. 3, or otherwise explain why certain Conditions are no longer applicable |
| 2.3 | Added a summary of industry activity since NRC approval of WCAP-17096-NP, Rev. 2 including the status of MRP-227, Rev. 1 and interim guidance for Westinghouse baffle-former bolts and guide cards |
| 2.4 | Added "Change Summary for WCAP-17096, Rev. 3" section |
| 3 | • Changed section title<br>• Added step to identify additional actions |
| 3.4 | Added clarification that the most appropriate inspection method has been determined based upon the relevant conditions |
| 3.6 | Added note regarding TLAA fulfilling fatigue evaluation |
| 3.7 | CE-ID: 3.2 (Ribs and Rings) was identified as inaccessible |
| 4.1 | Explained that content repeated from MRP-227 has been replaced with references to MRP-227 |
| 5.2 | • Made various changes to both Primary and Expansion components in Table 5-1 as necessary for consistency with MRP-227, Rev. 1<br>• Added "Vent Valve Assembly" component including the "Original Locking Devices", "Modified Locking Devices", and "Vent Valve Bodies"<br>• Added "Lower Grid Assembly" component and evaluation of the "Lower Grid Rib Section" and moved the "Shock Pad Bolts and their Locking Devices" from Expansion to Primary<br>• Deleted reference to Crystal River Unit 3 (CR-3) and associated components |



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**Table 2-1 Summary of Changes in WCAP-17096-NP, Rev. 3 relative to WCAP-17096-NP-A, Rev. 2**

| Section | Description of Change |
|---|---|
| | • Updated status on generic acceptance criteria in Table 5-1 as applicable |
| 6.2 | • Noted potential limitation on generic acceptance criteria to apply latest methodology<br>• Reordered and/or renamed Primary and Expansion components in Table 6-1 for CE design RVI and Table 6-2 for Westinghouse design RVI as necessary for consistency with MRP-227, Rev. 1<br>• Deleted "Upper Core Barrel Flange" CE component from Table 6-1 and "Outlet Nozzle Weld" Westinghouse component from Table 6-2<br>• Updated status on generic acceptance criteria in Table 6-1 and Table 6-2 as applicable |
| 7 | • Updated "Materials Reliability Program" references as applicable<br>• Added references related to MRP-227, Rev. 1 including RAI resolution and NRC Staff review<br>• Added interim guidance references related to baffle bolts, guide cards, and WCAP-17096 |
| Appendix A | • Removed content that was repeated from MRP-227 for each methodology<br>• Added, renamed, and changed Primary and Expansion components consistent with MRP-227, Rev. 1<br>• Various editorial and content type changes<br>➢ Refer directly to MRP-227 for appropriate content<br>• Other changes as indicated:<br>➢ A.1.1 (Core Clamping Items) – Addition of "Plenum Cover Support Ring', updated which units have performed measurements, added generic core clamping evaluation to Existing Documentation<br>➢ A.1.2 (Control Rod Guide Tube Spacer Castings) – Updated possible examination outcomes, included new wording about licensee analysis submittal, added reference to engineering evaluations and assessments to Existing Documentation<br>➢ A.1.3 (Vent Valve Assembly Vent Valve Retaining Rings) – Updated possible examination outcomes, included new wording about licensee analysis submittal, added reference to stress analyses to Existing Documentation<br>➢ A.1.4 (Vent Valve Assembly Original Locking Devices) – Addition of "Vent Valve Assembly Original Locking Devices," updated numbering of following sections accordingly<br>➢ A.1.5 (Vent Valve Assembly Modified Locking Devices) - Addition of "Vent Valve Assembly Modified Locking Devices," updated numbering of following sections accordingly<br>➢ A.1.6 (Upper Core Barrel Bolts and Locking Devices) – Changes to "UCB Bolts" methodology, removed wording in Existing Documentation section regarding future bolt failure rates and crack growth rates |



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**Table 2-1 Summary of Changes in WCAP-17096-NP, Rev. 3 relative to WCAP-17096-NP-A, Rev. 2**

| Section | Description of Change |
|---|---|
| | ➢ A.1.7 (Lower Core Barrel Bolts and Locking Devices) – Changes to "LCB Bolts" methodology, added wording in Existing Documentation section regarding existing unit specific analyses<br>➢ A.1.8 (Baffle-to-Former Bolts) – Changes to "Baffle-to-Former Bolts" methodology for consistency with PA-MSC-1596<br>➢ A.1.9 (Baffle Plates) – Updated CGR information in Existing Documentation section<br>➢ A.1.10 (Locking Devices and Locking Welds of Baffle-to-Former and Internal Baffle-to-Baffle Bolts) – No substantial changes<br>➢ A.1.11 (Flow Distributor Bolts and Locking Devices) – Changes to "FD Bolts" methodology, updated Existing Documentation<br>➢ A.1.12 (Guide Block Dowel Welds) – None<br>➢ A.1.13 (Shock Pad Bolts and Locking Devices) – Moved to Primary, updated component item function, included new wording about licensee analysis submittal, updated Existing Documentation, updated numbering of following section accordingly<br>➢ A.1.14 (Incore Monitoring Instrumentation Guide Tube Spiders and Welds) – Included new wording about licensee analysis submittal, updated Existing Documentation to reference existing unit-specific stress analyses<br>➢ A.2.1 (Upper Grid Fuel Assembly Support Pad Dowel Welds) – "Upper Grid Fuel Assembly Support Pad Dowel Welds" split from "Upper and Lower Grid Fuel Assembly Support Pad Dowel Welds" (moved to Section A.2.12), updated numbering of following sections accordingly<br>➢ A.2.2 (Vent Valve Assembly Vent Valve Bodies) – Addition of "Vent Valve Assembly Vent Valve Bodies," updated numbering of following sections accordingly<br>➢ A.2.3 (Upper Thermal Shield Bolts and Locking Devices) – Changes to "UTS Bolts" methodology, included new wording about licensee analysis submittal, updated Existing Documentation to reference existing unit-specific stress analyses<br>➢ A.2.4 (Surveillance Specimen Holder Tube Studs/Nuts of Bolts and Locking Devices (DB-1 only)) – Added note that applicability is only Davis Besse (D-B), updated Existing Documentation section<br>➢ A.2.5 (Core Barrel Cylinder) – Methodology now clarifies analysis in lieu of inspection, included new wording about licensee analysis submittal<br>➢ A.2.6 (Former Plates) - Methodology now clarifies analysis is in lieu of inspection, included new wording about licensee analysis submittal<br>➢ A.2.7 (Baffle-to-Baffle Bolts) - Methodology now clarifies analysis in lieu of inspection, included new wording about licensee analysis submittal<br>➢ A.2.8 (Core Barrel-to-Former Bolts) - Methodology now clarifies analysis in lieu of inspection, included new wording about licensee analysis submittal |



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| Table 2-1 Summary of Changes in WCAP-17096-NP, Rev. 3 relative to WCAP-17096-NP-A, Rev. 2 | |
|---|---|
| Section | Description of Change |
| | ➢ A.2.9 (Locking Devices for External Baffle-to-Baffle and Core Barrel-to-Former Bolts) – Methodology now clarifies analysis in lieu of inspection, included new wording about licensee analysis submittal<br>➢ A.2.10 (Lower Grid Fuel Assembly Support Pad Items) – None<br>➢ A.2.11 (Lower Grid Rib Section) - Addition of "Lower Grid Rib Section", updated numbering of following sections accordingly<br>➢ A.2.12 (Dowel to Lower Grid Fuel Assembly Support Pad Welds) – "Lower Grid Fuel Assembly Support Pad Dowel Welds" split from "Upper and Lower Grid Fuel Assembly Support Pad Dowel Welds" (moved to Section A.2.1), updated numbering of following section accordingly<br>➢ A.2.13 (Lower Thermal Shield Bolts and Locking Devices) – Included new wording about licensee analysis submittal, updated Existing Documentation to reference existing unit-specific stress analyses |
| Appendix B | New or updated flowcharts to match methodology changes for components in Appendix A |
| Appendix C | • Removed content that was repeated from MRP-277 from each methodology<br>• Added, renamed, and changed Primary and Expansion components consistent with MRP-227, Rev. 1<br>• Various editorial changes<br>• Other changes as indicated:<br>➢ CE-ID: 1 (Core Shroud Bolts) – None<br>➢ CE-ID: 1.1 (Core Support Column Bolts) – Added Additional Action(s)<br>➢ CE-ID: 1.2 (Barrel-Shroud Bolts) – None<br>➢ CE-ID: 2 (Core Shroud Plate-Former Plate Weld) – Added high fluence CGR, added option to use BWRVIP-158 flaw proximity rules, clarified that CGR may include K-independent basis, clarified safety factors, permitted alternate fracture toughness, and added Additional Action(s)<br>➢ CE-ID: 2.1 (Remaining Axial Welds) – Added high fluence CGR, added option to use BWRVIP-158 flaw proximity rules, clarified that CGR may include K-independent basis, clarified safety factors, permitted alternate fracture toughness, and added Additional Action(s)<br>➢ CE-ID: 3 (Shroud Plates) – Added high fluence CGR, added option to use BWRVIP-158 flaw proximity rules, clarified that CGR may include K-independent basis, clarified safety factors, permitted alternate fracture toughness, and added Additional Action(s)<br>➢ CE-ID: 3.1 (Remaining Axial Welds) – Added high fluence CGR, added option to use BWRVIP-158 flaw proximity rules, clarified that CGR may include K-independent basis, clarified safety factors, permitted alternate fracture toughness, and added Additional Action(s)<br>➢ CE-ID: 3.2 (Ribs and Rings) – Replaced flaw tolerance evaluation with functionality analysis<br>➢ CE-ID: 4 (Bolted CS Assembly) – Added Additional Action(s) |



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**Table 2-1 Summary of Changes in WCAP-17096-NP, Rev. 3 relative to WCAP-17096-NP-A, Rev. 2**

| Section | Description of Change |
|---|---|
| | ➢ CE-ID: 4a (Welded CS Assembly) – Added Additional Action(s)<br>➢ CE-ID: 5 (Upper Flange Weld) – Added option to use BWRVIP-158 flaw proximity rules, clarified that CGR may include K-independent basis, clarified safety factors, permitted alternate fracture toughness, and added Additional Action(s)<br>➢ CE-ID: 5.1 (Lower Girth Weld) – Added option to use BWRVIP-158 flaw proximity rules, clarified that CGR may include K-independent basis, clarified safety factors, permitted alternate fracture toughness, and added Additional Action(s)<br>➢ CE-ID: 5.2 (Upper Girth Weld) – Added option to use BWRVIP-158 flaw proximity rules, additional higher fluence cases, clarified that CGR may include K-independent basis, clarified safety factors, permitted alternate fracture toughness, and added Additional Action(s)<br>➢ CE-ID: 5.3 (Upper Axial Weld) – Added option to use BWRVIP-158 flaw proximity rules, high fluence CGR, clarified that CGR may include K-independent basis, clarified safety factors, permitted alternate fracture toughness, and added Additional Action(s)<br>➢ CE-ID: 5.4 (Lower Core Support Beams) – Added option to use BWRVIP-158 flaw proximity rules, clarified that CGR may include K-independent basis, clarified safety factors, permitted alternate fracture toughness, and added Additional Action(s)<br>➢ CE-ID: 6 (Middle Girth Weld) – Added option to use BWRVIP-158 flaw proximity rules, high fluence CGR, clarified that CGR may include K-independent basis, clarified safety factors, permitted alternate fracture toughness, and added Additional Action(s)<br>➢ CE-ID: 6.1 (Middle Axial Weld) – Added option to use BWRVIP-158 flaw proximity rules, high fluence CGR, clarified that CGR may include K-independent basis, clarified safety factors, permitted alternate fracture toughness, and added Additional Action(s)<br>➢ CE-ID: 6.2 (Lower Axial Weld) – Added option to use BWRVIP-158 flaw proximity rules, high fluence CGR, clarified that CGR may include K-independent basis, clarified safety factors, permitted alternate fracture toughness, and added Additional Action(s)<br>➢ CE-ID: 6.3 (Core Support Columns) – Added Additional Action(s)<br>➢ CE-ID: 7 (CSB Flexure Weld) – Added option to use BWRVIP-158 flaw proximity rules, clarified that CGR may include K-independent basis, clarified safety factors, permitted alternate fracture toughness, and added Additional Action(s)<br>➢ CE-ID: 8 (Core Support Plate) – Added Additional Action(s)<br>➢ CE-ID: 9 (Fuel Alignment Plate) – Added Additional Action(s)<br>➢ CE-ID: 10 (Instrument Guide Tubes) – None<br>➢ CE-ID: 10.1 (Remaining Instrument Guide Tubes) – None |



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| Table 2-1 Summary of Changes in WCAP-17096-NP, Rev. 3 relative to WCAP-17096-NP-A, Rev. 2 | |
|---|---|
| Section | Description of Change |
| | ➢ CE-ID: 11 (Deep Beams) – Added option to use BWRVIP-158 flaw proximity rules, additional higher fluence cases, clarified that CGR may include K-independent basis, clarified safety factors, and permitted alternate fracture toughness |
| Appendix D | Updated flowcharts to match methodology changes for components in Appendix C |
| Appendix E | • Removed content that was repeated from MRP-277 from each methodology<br>• Added, renamed, and changed Primary and Expansion components consistent with MRP-227, Rev. 1<br>• Various editorial changes<br>• Other changes as indicated:<br>➢ W-ID: 1 (Guide Plate(Cards)) – Updated methodology consistent with WCAP-17451-P, Rev. 2 changes<br>➢ W-ID: 2 (CRGT Lower Flange Welds) - Added Additional Action(s)<br>➢ W-ID: 2.1 (Remaining CRGT Lower Flange Welds) - Added Additional Action(s)<br>➢ W-ID: 2.2 (BMI Column Bodies) - None<br>➢ W-ID: 3 (Upper Flange Weld) – Added option to use BWRVIP-158 flaw proximity rules, clarified that CGR may include K-independent basis, clarified safety factors, permitted alternate fracture toughness, and added Additional Action(s)<br>➢ W-ID: 3.1 (Upper Girth Weld) - Added option to use BWRVIP-158 flaw proximity rules, added high fluence CGR, clarified that CGR may include K-independent basis, clarified safety factors, permitted alternate fracture toughness, and added Additional Action(s)<br>➢ W-ID: 3.2 (Upper Axial Weld) – Added option to use BWRVIP-158 flaw proximity rules, high fluence CGR, clarified that CGR may include K-independent basis, clarified safety factors, permitted alternate fracture toughness, and added Additional Action(s)<br>➢ W-ID: 3.3 (Lower Flange Weld) – Added option to use BWRVIP-158 flaw proximity rules, clarified that CGR may include K-independent basis, clarified safety factors, permitted alternate fracture toughness, and added Additional Action(s)<br>➢ W-ID: 3.4 (Lower Support Forging or Casting) – Added option to use BWRVIP-158 flaw proximity rules, and added Additional Action(s)<br>➢ W-ID: 4 (Lower Girth Weld) – Added option to use BWRVIP-158 flaw proximity rules, high fluence CGR, clarified that CGR may include K-independent basis, clarified safety factors, permitted alternate fracture toughness, and added Additional Action(s)<br>➢ W-ID: 4.1 (Upper Core Plate) - Added Additional Action(s)<br>➢ W-ID: 4.2 (Middle Axial Welds) – Added option to use BWRVIP-158 flaw proximity rules, high fluence CGR, clarified that CGR may include K- |



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**Table 2-1 Summary of Changes in WCAP-17096-NP, Rev. 3 relative to WCAP-17096-NP-A, Rev. 2**

| Section | Description of Change |
|---|---|
| | independent basis, clarified safety factors, permitted alternate fracture toughness, and added Additional Action(s)<br>➢ W-ID: 4.3 (Lower Axial Welds) - Added option to use BWRVIP-158 flaw proximity rules, high fluence CGR, clarified that CGR may include K-independent basis, clarified safety factors, permitted alternate fracture toughness, and Additional Action(s)<br>➢ W-ID: 4.4 (Lower Support Columns (Both Cast and Non-Cast)) - Added Additional Action(s)<br>➢ W-ID: 5 (Baffle-Edge Bolts) - Added Additional Action(s)<br>➢ W-ID: 6 (Baffle-Former Bolts) – Added interim guidance, option to use a probabilistic model, and added Additional Action(s)<br>➢ W-ID: 6.1 (Barrel-Former Bolts) - None<br>➢ W-ID: 6.2 (Lower Support Column Bolts) - Added Additional Action(s)<br>➢ W-ID: 7 (Baffle-Former Assembly - Assembly) – Added Additional Action(s)<br>➢ W-ID: 8 (Internals Hold Down Spring) - None<br>➢ W-ID: 9 (Thermal Shield Flexures) – Changed to “assessment” of fatigue, and added Additional Action(s) |
| Appendix F | Updated flowcharts to match methodology changes for components in Appendix E |



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| Table 2-2 Summary of Changes to WCAP-17096-NP-A, Rev. 3 associated with NRC RAI Responses | |
|---|---|
| Section | Description of Change |
| Introduction | Attached NRC transmittal letter and the Safety Evaluation for WCAP-17096-NP, Rev. 3 |
| 2.2 | The following changes were made based upon RAI responses:<br>• The Condition 1 Resolution for Group 1 RVI was updated to be consistent with response in RAIs 02 and 03<br>• The Condition 2 Resolution for Group 2 RVI was updated to be consistent with the response in RAI 04<br>• The Condition 3 Resolution for Group 4 RVI was updated to be consistent with the response in RAI 05 and 08<br>While not associated with any RAI response, the Condition 2 Resolution for Group 2 RVI was revised to acknowledge that Code Case N-889 was approved with NRC Conditions |
| 3 | A statement was added regarding the default assumptions for the development of methodologies per the response in Part (c) of RAI 13 |
| 3.3 | A statement about plant-specific applicability of generic ABPAs was removed per the resolution of Part (c) of RAI 07 |
| 3.5 | Added clarification that void swelling would be addressed as part of corrective action program per resolution of RAI 11 |
| 3.6 | • Section 3.6 regarding "Fatigue (Qualify by Evaluation)," and the associated footnotes were deleted per the response of RAI 08<br>• Section 3.7 regarding "Resolution by Analysis" was renumbered as Section 3.6 |
| 5.1 | Updated the statement regarding VT-3 acceptance criteria per the resolution of RAI 12 in Note 1 in Table 5-1 |
| 6.2 | • Deleted the bolted CE component methodologies CE-ID: 1, 1.1, and 1.2 from Table 6-1 per the resolution of Part (a) of RAI 07<br>• Corrected the numbering of the CE component methodologies CE-ID: 9, 10, 11, 11.1, and 12 from Table 6-1 to be consistent with MRP-227, Rev. 1-A per the response to RAI 08 |
| 7 | Updated Reference 36 to MRP-227, Rev. 1-A per the response to RAI 01 and added references 64-67 consistent with the response to RAI 04, Part (b) of RAI 07, and RAI 10 |
| Appendix A | • The following changes to component methodologies:<br>➢ A.1.1 (Core Clamping Items) –Updated the statement regarding VT-3 acceptance criteria per the resolution of RAI 12<br>➢ A.1.2 (Control Rod Guide Tube Spacer Castings) - Updated the statement regarding VT-3 acceptance criteria per the resolution of RAI 12<br>➢ A.1.3 (Vent Valve Assembly Vent Valve Retaining Rings) –Updated the statement regarding VT-3 acceptance criteria per the resolution of RAI 12 |



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| Table 2-2 | Summary of Changes to WCAP-17096-NP-A, Rev. 3 associated with NRC RAI Responses |
|---|---|
| Section | Description of Change |
| | ➢ A.1.6 (Upper Core Barrel Bolts and Locking Devices) – Updated the statement regarding VT-3 acceptance criteria for the locking devices per the resolution of RAI 12<br>➢ A.1.7 (Lower Core Barrel Bolts and Locking Devices) – Updated the statement regarding VT-3 acceptance criteria for the locking devices per the resolution of RAI 12<br>➢ A.1.9 (Baffle Plates) – Updated the statement regarding VT-3 acceptance criteria per the resolution of RAI 12, and added submittal requirement per resolution of RAI 03<br>➢ A.1.10 (Locking Devices and Locking Welds of Baffle-to-Former and Internal Baffle-to-Baffle Bolts) – Updated the statement regarding VT-3 acceptance criteria per the resolution of RAI 12<br>➢ A.1.11 (Flow Distributor Bolts and Locking Devices) – Updated the statement regarding VT-3 acceptance criteria for the locking devices per the resolution of RAI 12, and updated Existing Documentation<br>➢ A.1.12 (Guide Block Dowel Welds) – Updated the statement regarding VT-3 acceptance criteria per the resolution of RAI 12<br>➢ A.1.13 (Shock Pad Bolts and Locking Devices) – Updated the statement regarding VT-3 acceptance criteria for the locking devices per the resolution of RAI 12<br>➢ A.1.14 (Incore Monitoring Instrumentation Guide Tube Spiders and Welds) - Updated the statement regarding VT-3 acceptance criteria per the resolution of RAI 12<br>➢ A.2.1 (Upper Grid Fuel Assembly Support Pad Dowel Welds) - Updated the statement regarding VT-3 acceptance criteria per the resolution of RAI 12<br>➢ A.2.3 (Upper Thermal Shield Bolts and Locking Devices) – Updated the statement regarding VT-3 acceptance criteria for the locking devices per the resolution of RAI 12<br>➢ A.2.4 (Surveillance Specimen Holder Tube Studs/Nuts of Bolts and Locking Devices (DB-1 only)) – Updated the statement regarding VT-3 acceptance criteria for the locking devices per the resolution of RAI 12<br>➢ A.2.5 (Core Barrel Cylinder) – Added text to the submittal requirement per the resolution of RAI 02<br>➢ A.2.6 (Former Plates) - Added text to the submittal requirement per the resolution of RAI 02<br>➢ A.2.7 (Baffle-to-Baffle Bolts) - Added text to the submittal requirement per the resolution of RAI 02<br>➢ A.2.8 (Core Barrel-to-Former Bolts) - Added text to the submittal requirement per the resolution of RAI 02<br>➢ A.2.9 (Locking Devices for External Baffle-to-Baffle and Core Barrel-to-Former Bolts) - Added text to the submittal requirement per the resolution of RAI 02 |



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| Table 2-2 Summary of Changes to WCAP-17096-NP-A, Rev. 3 associated with NRC RAI Responses | |
|---|---|
| Section | Description of Change |
| | ➢ A.2.10 (Lower Grid Fuel Assembly Support Pad Items) – Updated the statement regarding VT-3 acceptance criteria per the resolution of RAI 12<br>➢ A.2.12 (Dowel to Lower Grid Fuel Assembly Support Pad Welds) – Updated the statement regarding VT-3 acceptance criteria per the resolution of RAI 12<br>➢ A.2.13 (Lower Thermal Shield Bolts and Locking Devices) – Updated the statement regarding VT-3 acceptance criteria for the locking devices per the resolution of RAI 12 |
| Appendix B | No changes |
| Appendix C | • The following changes to component methodologies:<br>➢ CE-ID: 1 (Core Shroud Bolts), CE-ID: 1.1 (Core Support Column Bolts), and CE-ID: 1.2 (Barrel-Shroud Bolts) – Methodologies removed per resolution of Part (b) of RAI 07<br>➢ CE-ID: 2 (Core Shroud Plate-Former Plate Weld) – Added CGR criteria per resolution of RAI 04, updated for NRC approval of Code Case N-889 with Conditions, and identified the source of high dose fracture toughness values per RAI 10<br>➢ CE-ID: 2.1 (Remaining Axial Welds) – Added CGR criteria per resolution of RAI 04, updated for NRC approval of Code Case N-889 with Conditions, and identified the source of high dose fracture toughness values per RAI 10<br>➢ CE-ID: 3 (Shroud Plates) – Added CGR criteria per resolution of RAI 04, updated for NRC approval of Code Case N-889 with Conditions, and identified the source of high dose fracture toughness values per RAI 10<br>➢ CE-ID: 3.1 (Remaining Axial Welds) – Added CGR criteria per resolution of RAI 04, updated for NRC approval of Code Case N-889 with Conditions, and identified the source of high dose fracture toughness values per RAI 10<br>➢ CE-ID: 3.2 (Ribs and Rings) – Added an Additional Action(s) per the resolution of RAI 05.<br>➢ CE-ID: 5 (Upper Flange Weld) – Added CGR criteria per resolution of RAI 04, and updated for NRC approval of Code Case N-889 with Conditions<br>➢ CE-ID: 5.1 (Lower Girth Weld) – Added CGR criteria per resolution of RAI 04, and updated for NRC approval of Code Case N-889 with Conditions<br>➢ CE-ID: 5.2 (Upper Girth Weld) – Added CGR criteria per resolution of RAI 04, and updated for NRC approval of Code Case N-889 with Conditions, and identified the source of high dose fracture toughness values per RAI 10<br>➢ CE-ID: 5.3 (Upper Axial Weld) – Added CGR criteria per resolution of RAI 04, updated for NRC approval of Code Case N-889 with Conditions, and identified the source of high dose fracture toughness values per RAI 10<br>➢ CE-ID: 5.4 (Lower Core Support Beams) – Added CGR criteria per resolution of RAI 04, and updated for NRC approval of Code Case N-889 with Conditions |



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| Table 2-2 | Summary of Changes to WCAP-17096-NP-A, Rev. 3 associated with NRC RAI Responses |
|---|---|
| Section | Description of Change |
| | ➢ CE-ID: 6 (Middle Girth Weld) – Added CGR criteria per resolution of RAI 04, updated for NRC approval of Code Case N-889 with Conditions, and identified the source of high dose fracture toughness values per RAI 10<br>➢ CE-ID: 6.1 (Middle Axial Weld) – Added CGR criteria per resolution of RAI 04, updated for NRC approval of Code Case N-889 with Conditions, and identified the source of high dose fracture toughness values per RAI 10<br>➢ CE-ID: 6.2 (Lower Axial Weld) – Added CGR criteria per resolution of RAI 04, updated for NRC approval of Code Case N-889 with Conditions, and identified the source of high dose fracture toughness values per RAI 10<br>➢ CE-ID: 7 (CSB Flexure Weld) – Added CGR criteria per resolution of RAI 04, updated for NRC approval of Code Case N-889 with Conditions, and removed the discussion of fatigue and SCC screening from inspection method per response to Part (b) of RAI 08<br>➢ CE-ID: 9 (Core Support Plate) – Section re-numbered per resolution of RAI 08, and removed the discussion of fatigue screening from inspection method per response to Part (b) of RAI 08<br>➢ CE-ID: 10 (Fuel Alignment Plate) – Section re-numbered per resolution of RAI 08, and removed the discussion of fatigue screening from inspection method per response to Part (b) of RAI 08<br>➢ CE-ID: 11 (Instrument Guide Tubes) – Section re-numbered per resolution of RAI 08<br>➢ CE-ID: 11.1 (Remaining Instrument Guide Tubes) – Section re-numbered per resolution of RAI 08<br>➢ CE-ID: 12 (Deep Beams) – Section re-numbered per resolution of RAI 08, added CGR criteria per resolution of RAI 04, updated for NRC approval of Code Case N-889 with Conditions, and identified the source of high dose fracture toughness values per RAI 10 |
| Appendix D | • Updated flowcharts to match methodology changes for components in Appendix C<br>➢ CE-ID: 1 (Core Shroud Bolts), CE-ID: 1.1 (Core Support Column Bolts), and CE-ID: 1.2 (Barrel-Shroud Bolts – Methodologies removed per resolution of Part (a) of RAI 07<br>➢ CE-ID: 9, 10, 11, 11.1, and 12 re-numbered per resolution of RAI 08 |
| Appendix E | • The following changes to the component methodologies:<br>➢ W-ID: 1 (Guide Plate(Cards)) – Corrected the content of the 3rd bulleted item in the guide card data requirements per resolution of RAI 14<br>➢ W-ID: 3 (Upper Flange Weld) – Added CGR criteria per resolution of RAI 04, and updated for NRC approval of Code Case N-889 with Conditions<br>➢ W-ID: 3.1 (Upper Girth Weld) - Added CGR criteria per resolution of RAI 04, updated for NRC approval of Code Case N-889 with Conditions, and identified the source of high dose fracture toughness values per RAI 10 |



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| Table 2-2 | Summary of Changes to WCAP-17096-NP-A, Rev. 3 associated with NRC RAI Responses |
|---|---|
| Section | Description of Change |
| | ➢ W-ID: 3.2 (Upper Axial Weld) – Added CGR criteria per resolution of RAI 04, updated for NRC approval of Code Case N-889 with Conditions, and identified the source of high dose fracture toughness values per RAI 10<br>➢ W-ID: 3.3 (Lower Flange Weld) –Added CGR criteria per resolution of RAI 04, updated for NRC approval of Code Case N-889 with Conditions<br>➢ W-ID: 4 (Lower Girth Weld) –Added CGR criteria per resolution of RAI 04, updated for NRC approval of Code Case N-889 with Conditions, and identified the source of high dose fracture toughness values per RAI 10<br>➢ W-ID: 4.2 (Middle Axial Welds) – Added CGR criteria per resolution of RAI 04, updated for NRC approval of Code Case N-889 with Conditions, and identified the source of high dose fracture toughness values per RAI 10<br>➢ W-ID: 4.3 (Lower Axial Welds) - Added CGR criteria per resolution of RAI 04, updated for NRC approval of Code Case N-889 with Conditions, and identified the source of high dose fracture toughness values per RAI 10<br>➢ W-ID: 6 (Baffle-Former Bolts) – Acknowledged that the supplemental ABPA methodology is available per Part (b) of RAI 07, clarification on how untestable bolts will be evaluated per Part (e) of RAI 13, clarification that re-inspection longer than 10 years are considered a deviation per Part (f) of RAI 13, and clarification about the applicability of generic ABPAs per Part (c) of RAI 07<br>➢ W-ID: 6.1 (Barrel-Former Bolts) - Acknowledged that the supplemental ABPA methodology is available per Part (b) of RAI 07 |
| Appendix F | No changes |
| Appendix G | Added correspondence and RAI responses associated with NRC-approved version |



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# 3 EVALUATION PROCESS

The goal of this effort is to define the process to be used in the degradation evaluation procedures and the necessary data requirements to perform the evaluation of each Primary and Expansion component such that engineering organizations follow a consistent approach that is documented and approved. The guidelines and methodologies are listed in outline format and illustrated in a flowchart showing the evaluation and disposition options for the relevant condition.

Note that the following default assumptions have been applied in the development of the methodologies contained herein:

- The degradation of each component is assumed to be independent of the degradation of other components (i.e. no interaction), and the loading applied in the methodologies assumes the other components are functioning as designed.

Both Framatome and Westinghouse assembled teams of experts to develop recommendations. Although the procedural details and reporting format of these efforts are unique to the vendors, the basic process followed the same general steps:

1. Review MRP-227 degradation modes and inspection recommendations.
2. Determine component function.
3. List potential inspection outcomes and observable effects.
4. Identify potential failure mechanisms and effects.
5. Outline methodology to evaluate potential inspection observations.
6. Determine data requirements for inspection.
7. Consider potential for vendor-specific generic analysis of the components.
8. Identify additional actions.

The form and structure of the evaluation methodology is determined by the type of inspection recommended.

## 3.1 PHYSICAL MEASUREMENTS

Physical measurements are employed to characterize changes in component dimension. For example, measurements of internals hold down spring height are used to evaluate loss of core hold-down forces due to wear or stress relaxation. In this case, the required hold-down forces are a design requirement and generic or plant-specific acceptance criteria should be established prior to the inspection.

No specific action to establish general acceptance criteria was required under this task. The required methodologies simply note the existence of relevant design requirements for the affected components.

## 3.2 GENERAL CONDITION

There is a heavy reliance on general condition monitoring within the I&E recommendations. Although these recommendations refer to the VT-3 level visual inspections, the guidelines generally provide a description of the expected degradation. Under the VT-3 procedure, any observation of degradation is



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considered to be a relevant condition and is reported to engineering for evaluation. Generally, engineering will review the observation to either confirm the relevant condition or disposition it as a visual anomaly.

Due to the qualitative nature of the VT-3 examination, it is difficult to define quantitative evaluation and acceptance criteria. In some cases, VT-3 examinations have been specified for redundant components, where multiple failures are required to impair the functionality of the system. In other cases, VT-3 has been specified where the first signs of degradation are expected to be visual (e.g., wear). In any case, it is prudent engineering practice to anticipate the range of possible visual observations and define resolution strategies.

A process identifying potential relevant conditions arising from each recommended VT-3 examination and defining resolution strategies is required for each VT-3 examination. This process may take the form of a failure modes and effects analysis (FMEA).

## 3.3 ULTRASONIC TESTING (BOLTS)

Within the I&E Guidelines, ultrasonic testing (UT) examinations are conducted to identify failed bolts. All bolts with positive indications of cracking are assumed to be failed. The bolting systems in the internals are generally highly redundant. Acceptance criteria are based on acceptable bolting patterns that guarantee structural stability through both normal operation and design basis transients. To establish an appropriate inspection interval, the current distribution of unfailed bolts must contain sufficient margin to demonstrate that the number of anticipated failures will not cause the distribution to fall below the minimum pattern. This will require either a historical analysis of bolt failure rates or a detailed model of bolt failure mechanisms.

The use of acceptable bolting patterns as acceptance criteria that allow individual bolt failures is established in the industry. The PWROG (from prior Westinghouse Owners Group [WOG] efforts) has developed acceptable bolting patterns for baffle-to-former bolting for Westinghouse reactor internals designs. The PWROG has supported development of similar strategies for core barrel bolt inspections in Babcock & Wilcox (B&W) plants. The methodology for performing an acceptable bolting pattern or similar strategy is beyond the scope of the current task.

## 3.4 VISUAL CRACKING

Visual examinations to identify cracking are generally recommended where irradiation-assisted stress corrosion cracking (IASCC), intergranular stress corrosion cracking (IGSCC), stress corrosion cracking (SCC), or fatigue is identified as the cracking mechanism. The initial visual examinations in the B&W plants are all based on VT-3 requirements. The VT-3 exams were deemed to be adequate because the structures are relatively flaw tolerant. Appropriate follow-on actions might include EVT-1, UT, or eddy-current testing (ET) exams to determine flaw size. These options would be identified as part of the recommended procedure for resolution of the original VT-3 observation. Consistent with the current state-of-the-art within the Boiling Water Reactor Vessel and Internals Project (BWRVIP), EVT-1 has been identified as the appropriate visual examination procedure for several of the Combustion Engineering (CE) and Westinghouse components. The EVT-1 examination can produce flaw size information that would lead directly to a fracture mechanics evaluation. For each component, the most appropriate visual



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inspection method has been determined (e.g., EVT-1 or VT-3) based upon the relevant conditions (i.e. flaw tolerance, inspectability, etc.). For components where a visual examination is identified, the specific technique is specified in MRP-227. UT or ET may also be used as enhanced or supplemental examinations, where appropriate.

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

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

## 3.5 VISUAL OTHER

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

These examinations are meant to provide an early warning of swelling in the structure. A functionality evaluation was performed in MRP-230 [6], which shows that gap openings may indicate void swelling. Additional sensitivity studies could be performed to relate the swelling level to the predicted distortion and gap opening in the structure, which would be used to evaluate the current condition, predict future effects, and determine aging management strategies. Void swelling, if identified, will be addressed in the licensee's corrective action program.

## 3.6 RESOLUTION BY ANALYSIS

Three Expansion component items in the B&W designs and one component in the CE design have been designated for resolution by analysis[1]. Inspection of these three components is considered to be impractical due to issues of accessibility. Therefore, should concerns about the integrity of these components be triggered by observations in the associated Primary components, no inspection is required. Resolution would require either detailed analysis or replacement.

Therefore, inspection acceptance criteria for the three Expansion components in the B&W design and the one component in the CE design are not included in this task.

[1] The referenced three B&W component items are: 1) the core barrel cylinder and welds, 2) the former plates, and 3) the bolting (core barrel-to-former bolts, internal (accessible, but currently unable to be ultrasonically inspected) and external baffle-to-baffle bolts, and associated locking devices), and the referenced CE component is the ribs and rings.



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# 4 RECOMMENDATION FORMAT

Recommendations for each Primary and Expansion item were prepared by Westinghouse and Framatome after consulting with appropriate experts and expert panels. The basic requirements for this study were mutually agreed upon between the vendors and defined in the original project authorization. The internal processes employed to complete these studies were specific to the vendor. The results of the Westinghouse and Framatome analyses are provided in Appendix A (B&W plants), Appendix C (CE plants), and Appendix E (Westinghouse plants). Although there are superficial differences in the format in which the results are presented by each vendor, the underlying structure of the information is the same.

## 4.1 IDENTIFICATION AND EXAMINATION RECOMMENDATIONS

Component identification and inspection recommendations for both the Primary and Expansion component items are provided in MRP-227. In Revision 3 of this report, content in the appendices that originates from, and resides in, MRP-227 was removed. In its place, a reference back to MRP-227 was added so that the content in this document does not become outdated should MRP-227 be revised.

## 4.2 COMPONENT FUNCTION

To facilitate consideration of consequence of failure, both vendor processes include a brief summary of the component function. More extensive function descriptions for the reactor internals components were originally compiled in support of the Consequence Analysis performed for the Issue Management Tables. These summaries are included in MRP-156 [11], MRP-157 [51], and MRP-205 [50]. The component function summaries provided in Appendices A, C, and E are meant only to provide perspective for the analysis recommendations.

## 4.3 INSPECTION OUTCOMES

The inspection recommendations provided in MRP-227 are based on the identification of specific aging degradation mechanisms. The expert teams were asked to consider the information potentially generated by the inspection. The Framatome evaluations are summarized in Appendix A under the headings “Observable Effects” and “Possible Outcomes.” The Westinghouse evaluations are summarized in Appendices C and E under the headings “Observable Effect,” “Failure Mechanism,” “Failure Effect,” and “Failure Criteria.”

## 4.4 METHODOLOGY AND DATA REQUIREMENTS

The objective of this study was to define methodologies and data requirements for analysis and acceptance of degraded components identified in the MRP-227 recommended inspection. The Framatome format provides this information under a single heading. The Westinghouse format divides the methodology into four separate subheadings: “Goal,” “Data Requirements,” “Analysis,” and “Acceptance Criteria.” For the Westinghouse sections, unless otherwise indicated by specific terminology, “operating loads” refers to any loads generated under normal operating conditions.



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## 4.5 RECOMMENDED APPROACH

In general, implementation of the methodology and data requirement recommendations can require a combination of plant-specific and cooperative actions. Because the MRP-227 recommendations tend to be very design specific, generic actions that apply to the entire PWR fleet are not expected. However, it may be possible to define cooperative activities relevant to rational subgroups such as plants designed by a single vendor.



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# 5 B&W PLANT DESIGN RESULTS

## 5.1 METHODOLOGIES

A summary of the guidelines and methodology for determining acceptance criteria and the needed data requirements suggested by the expert panel for each of the Primary and Expansion component items identified in MRP-227 for the operating B&W-design reactor vessel internals is provided in Appendix A.

Appendix B provides flowcharts illustrating the evaluation path and potential disposition options for relevant inspection conditions for each of the Primary and Expansion component items identified in MRP-227.

## 5.2 GENERIC ACCEPTANCE CRITERIA

Each of the Primary and Expansion component items were evaluated to determine those for which analyses would be practicable to develop generic acceptance criteria. Differing unit loads, transients, materials, etc. were considered in identifying those component items that could be analyzed on a generic basis. The Framatome analysis indicates actions that might be used to define nondestructive examination (NDE) acceptance standards and actions that could support analytical evaluations. Table 5-1 that follows provides the results of this effort.



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**Table 5-1 Applicability of Potential Generic Acceptance Criteria for B&W-Design Primary and Expansion Component Items**

| Component Item | Develop Generic Acceptance Criteria?(1) | | Comments |
|---|---|---|---|
| | Analytical | NDE | |
| **Primary Items** | | | |
| **Plenum Cover Assembly & Core Support Shield Assembly**<br>Plenum cover weldment rib pads<br>Plenum cover support flange<br>Plenum cover support ring<br>CSS top flange | Yes | Yes | A generic core clamping evaluation can be performed, but unit-specific is recommended. |
| **Control Rod Guide Tube Assembly**<br>CRGT spacer castings | N/A | Yes | A generic degraded casting evaluation can be performed.<br>Reactivity analyses, if necessary, are dependent upon fuel loading and must be performed on a unit-specific basis. |
| **Vent Valve Assembly**<br>Vent valve top retaining ring<br>Vent valve bottom retaining ring | Yes | Yes | A generic vent valve retaining ring evaluation can be performed, but unit-specific efforts may provide additional margin (particularly for DB-1). |
| **Vent Valve Assembly**<br>Original locking devices (pressure plate, spring retainer, spring, U-cover) | No | Yes | Given the unique nature of observed degradation (e.g., contact during CSA outage movements), evaluation as to replace or leave as-is is on a case-by-case basis. |
| **Vent Valve Assembly**<br>Original locking devices (key ring, pin) | No | Yes | Given the unique nature of observed degradation (e.g., contact during CSA outage movements), evaluation as to replace or leave as-is is on a case-by-case basis. |
| **Vent Valve Assembly**<br>Modified locking devices (bolt locking cup, jackscrew locking cup, bolted block) | No | Yes | Given the unique nature of observed degradation (e.g., contact during CSA outage movements), evaluation as to replace or leave as-is is on a case-by-case basis. |
| **Core Support Shield Assembly**<br>Upper core barrel (UCB) bolts and their locking devices | No | Yes | Due to the variations in bolt materials used and loadings among the units, unit-specific analytical efforts are required. The generic efforts have already been completed in PWROG work. |



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| Table 5-1 (cont.) Applicability of Potential Generic Acceptance Criteria for B&W-Design Primary and Expansion Component Items | | | |
|---|---|---|---|
| **Component Item** | **Develop Generic Acceptance Criteria?(1)** | | **Comments** |
| | **Analytical** | **NDE** | |
| **Core Barrel Assembly**<br>Lower core barrel (LCB) bolts and their locking devices | No | Yes | Due to the variations in bolt materials used and loadings among the units, unit-specific analytical efforts are required. The generic efforts have already been completed in PWROG work. |
| **Core Barrel Assembly**<br>Baffle-to-former bolts | Yes | Yes | A generic safety assessment for BF bolt cracking could be performed. |
| **Core Barrel Assembly**<br>Baffle plates | Yes | Yes | A generic safety assessment for baffle plate cracking could be performed. |
| **Core Barrel Assembly**<br>Locking devices, including locking welds, of baffle-to-former bolts and internal baffle-to-baffle bolts | N/A | Yes | There are two slightly different designs that need to be analyzed, but a generic assessment could be performed. |
| **Flow Distributor Assembly**<br>Flow distributor (FD) bolts and their locking devices | Yes | Yes | Analytical efforts for the FD bolts could be performed on a generic basis (for all units except TMI-1), although unit-specific analyses could decrease the conservatism for some units. |
| **Lower Grid Assembly**<br>Alloy X-750 dowel-to-guide block welds | N/A | Yes | A generic evaluation of the minimum number of needed guide blocks can be performed (for all units except DB-1). |
| **Lower Grid Assembly**<br>Shock pad bolts and their locking devices | No | No | TMI-1 only, so unit-specific analysis is required. |
| **Incore Monitoring Instrumentation (IMI) Guide Tube Assembly**<br>IMI guide tube spiders<br>IMI guide tube spider-to-lower grid rib section welds | Yes | Yes | A generic analysis for the IMI guide tube spiders and spider-to-lower grid rib section welds could be performed. |
| **Expansion Items** | | | |
| **Upper Grid Assembly**<br>Alloy X-750 dowel-to-upper grid fuel assembly support pad welds | N/A | Yes | A generic evaluation (for all units except DB-1) of the consequences of failure of the weld, and if necessary, any additional examination required to confirm functional acceptability can be performed. |



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| Table 5-1 (cont.) Applicability of Potential Generic Acceptance Criteria for B&W-Design Primary and Expansion Component Items | | | |
|---|---|---|---|
| **Component Item** | **Develop Generic Acceptance Criteria?(1)** | | **Comments** |
| | **Analytical** | **NDE** | |
| **Vent Valve Assembly**<br>Vent valve bodies | No | Yes | Given the unique nature of observed degradation (e.g., contact during CSA outage movements), evaluation as to replace or leave as-is may need to be on a case-by-case basis. |
| **Core Barrel Assembly**<br>Upper thermal shield (UTS) bolts and their locking devices | Yes | Yes | Analytical efforts for the UTS bolt failures could be performed on a generic basis (for all units except TMI-1), although use of unit-specific loadings could reduce the conservatism for some units. |
| **Core Barrel Assembly**<br>Surveillance specimen holder tube (SSHT) bolts and their locking devices | No | No | DB-1 only, so unit-specific analyses are required. |
| **Core Barrel Assembly**<br>Core barrel cylinder (including vertical and circumferential seam welds)<br>Former plates | Yes | N/A | The efforts have already been completed. |
| **Core Barrel Assembly**<br>Baffle-to-baffle bolts<br>Core barrel-to-former bolts | Yes | N/A | The efforts have already been completed. |
| **Core Barrel Assembly**<br>Locking devices, including locking welds, for the external baffle-to-baffle bolts and core barrel-to-former bolts | N/A | N/A | The efforts have already been completed. |
| **Lower Grid Assembly**<br>Lower grid fuel assembly support pad items: pad, pad-to-rib section welds, Alloy X-750 dowel, cap screw, and their locking welds<br>(Note: The pads, dowels, and cap screws are included because of IE of the welds.) | N/A | Yes | An evaluation of the lower grid assembly items of the consequences of failure for each item and, if necessary, any additional examination required to confirm functional acceptability could be performed generically. |
| **Lower Grid Assembly**<br>Lower grid rib section | Yes | N/A | A generic analysis for the lower grid rib section could be performed. |



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| Table 5-1 (cont.) Applicability of Potential Generic Acceptance Criteria for B&W-Design Primary and Expansion Component Items | | | |
|---|---|---|---|
| **Component Item** | **Develop Generic Acceptance Criteria?(1)** | | **Comments** |
| | **Analytical** | **NDE** | |
| **Lower Grid Assembly**<br>Alloy X-750 dowel-to-lower grid fuel assembly support pad welds | N/A | Yes | A generic evaluation (with the exception of DB-1) of the consequences of failure of the weld, and if necessary, any additional examination required to confirm functional acceptability can be performed. |
| **Lower Grid Assembly**<br>Lower thermal shield studs/nuts or bolts (LTS) and their locking devices | Yes | Yes | Generic efforts can be performed for 1) stud/nut and 2) bolt materials, although use of unit-specific analyses could decrease the conservatism for some units. |

**Notes:**
(1) Analytical efforts include finite element analysis or fracture mechanics analysis. Relevant indications for the B&W items are defined in Section 5 of MRP-227-1A. These relevant conditions constitute discontinuities or imperfections (e.g., loose or missing parts, distortion, corrosion, wear, etc.), which are readily detectable with the VT-3 technique.



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# 6 COMBUSTION ENGINEERING AND WESTINGHOUSE PLANT DESIGN RESULTS

## 6.1 METHODOLOGIES

Datasheets outlining the guidelines and methodology for determining acceptance criteria and the needed data requirements suggested by the expert panel for each of the Primary and Expansion component items in CE and Westinghouse plants are provided in Appendices C and E. CE recommendations are contained in Appendix C. Westinghouse plant recommendations are contained in Appendix E.

Appendices D and F provide flowcharts illustrating the evaluation path and potential disposition options for relevant inspection conditions for each of the Primary and Expansion component items identified in MRP-227.

## 6.2 GENERIC ACCEPTANCE CRITERIA

Comments on the analysis approach for each component are included in the final section of the datasheets included in Appendices C and E. These recommendations, which include any actions that might be taken on a generic basis, are summarized in Tables 6-1 and 6-2. Note that the generic PWROG evaluations as discussed in the following are based upon previous versions of methodologies, which could require an update to be consistent with the most recent methodologies as defined herein.



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**Table 6-1 Applicability of Potential Generic Acceptance Criteria for CE-Design Primary and Expansion Component Items**

| CE Component | Approach |
|---|---|
| CE-ID: 2 Core Shroud Assembly (Welded) – Core Shroud Plate-Former Plate Weld | Limit load, EPFM, or LEFM analysis.<br>PWROG generic flaw evaluation complete. |
| CE-ID: 2.1 Core Shroud Assembly (Welded) - Remaining Axial Welds | Limit load, EPFM, or LEFM analysis. |
| CE-ID: 3 Core Shroud Assembly (Welded – Shroud Plates | Limit load, EPFM, or LEFM analysis.<br>PWROG generic flaw evaluation complete. |
| CE-ID: 3.1 Core Shroud Assembly (Welded) – Remaining Axial Welds | Limit load, EPFM, or LEFM analysis. |
| CE-ID: 3.2 Core Shroud Assembly (Welded) – Ribs and Rings | Inaccessible for inspection. Functionality analysis |
| CE-ID: 4 Core Shroud Assembly (Bolted) – Assembly | No generic effort required. Only one plant is affected. |
| CE-ID: 4a Core Shroud Assembly (Welded) – Assembly | Generic efforts to support inspection.<br>- Extension of MRP model to look at relationship between swelling and deformation at seam.<br>- Guideline for issues to be addressed in plant-specific FMEA. |
| CE-ID: 5 Core Support Barrel Assembly – Upper Flange Weld | Limit load or LEFM analysis.<br>PWROG generic flaw evaluation complete. |
| CE-ID: 5.1 Core Support Barrel Assembly – Lower Girth Weld | Limit load or LEFM analysis.<br>PWROG generic flaw evaluation complete. |
| CE-ID: 5.2 Core Support Barrel Assembly – Upper Girth Weld | Limit load or LEFM analysis.<br>PWROG generic flaw evaluation complete. |
| CE-ID: 5.3 Core Support Barrel Assembly – Upper Axial Weld | Limit load, EPFM, or LEFM analysis.<br>PWROG generic flaw evaluation complete. |
| CE-ID: 5.4 Lower Support Structure – Lower Core Support Beams | Limit load or LEFM analysis. |
| CE-ID: 6 Core Support Barrel Assembly – Middle Girth Weld | Limit load, EPFM, or LEFM analysis.<br>PWROG generic flaw evaluation complete. |
| CE-ID: 6.1 Core Support Barrel Assembly – Middle Axial Weld | Limit load, EPFM, or LEFM analysis.<br>PWROG generic flaw evaluation complete. |
| CE-ID: 6.2 Core Support Barrel Assembly – Lower Axial Weld | Limit load, EPFM, or LEFM analysis.<br>PWROG generic flaw evaluation complete. |
| CE-ID: 6.3 Lower Support Structure – Core Support Columns | Pattern analysis specific to the lower support structure design and loads.<br>PWROG functionality analysis complete. |



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**Table 6-1 Applicability of Potential Generic Acceptance Criteria for CE-Design Primary and Expansion Component Items**

| CE Component | Approach |
|---|---|
| CE-ID: 7 Core Support Barrel Assembly – CSB Flexure Weld | Determine allowable flaw length using fracture mechanics evaluation.<br>PWROG generic flaw evaluation complete. |
| CE-ID: 9 Lower Support Structure – Core Support Plate | FEA (plant specific) |
| CE-ID: 10 Upper Internals Assembly – Fuel Alignment Plate | FEA (plant specific) |
| CE-ID: 11 Control Element Assembly –Instrument Guide Tubes | Pass/Fail inspection with established minimum number of instrumentation tubes. Based directly on plant specifications. |
| CE-ID: 11.1 Control Element Assembly – Remaining Instrument Guide Tubes | Pass/Fail inspection with established minimum number of instrumentation tubes. Based directly on plant specifications. |
| CE-ID: 12 Lower Support Structure – Deep Beams | Determine allowable flaw length using fracture mechanics evaluation. |



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**Table 6-2 Applicability of Potential Generic Acceptance Criteria for Westinghouse-Design Primary and Expansion Component Items**

| Westinghouse Component | Approach |
|---|---|
| W-ID: 1 Control Rod Guide Tube Assembly – Guide Plates (Cards) | Generic work completed in existing PWROG program WCAP-17451-P Rev. 2. |
| W-ID: 2 Control Rod Guide Tube Assembly – Lower Flange Welds | Plant-specific analysis due to large variety of sizes and designs. There may be some potential for smaller plant groupings. |
| W-ID: 2.1 Control Rod Guide Tube Assembly – Remaining CRGT Lower Flange Welds | Plant-specific analysis due to large variety of sizes and designs. There may be some potential for smaller plant groupings. |
| W-ID: 2.2 Bottom-Mounted Instrumentation (BMI) System – BMI Column Bodies | Pass/Fail inspection with established minimum number of instrumentation tubes. Based directly on plant technical specifications. |
| W-ID: 3 Core Barrel Assembly – Upper Flange Weld | Limit load or LEFM analysis.<br>PWROG generic flaw evaluation complete. |
| W-ID: 3.1 Core Barrel Assembly –Upper Girth Weld | Limit load, EPFM, or LEFM analysis.<br>PWROG generic flaw evaluation complete. |
| W-ID: 3.2 Core Barrel Assembly – Upper Axial Weld | Limit load, EPFM, or LEFM analysis.<br>PWROG generic flaw evaluation complete. |
| W-ID: 3.3 Core Barrel Assembly – Lower Flange Weld | Limit load or LEFM analysis.<br>PWROG generic flaw evaluation complete. |
| W-ID: 3.4 Lower Internals Assembly – Lower Support Forging or Castings | FEA (plant specific) |
| W-ID: 4 Core Barrel Assembly –Lower Girth Weld | Limit load, EPFM, or LEFM analysis.<br>PWROG generic flaw evaluation complete. |
| W-ID: 4.1 Upper Internals Assembly – Upper Core Plate | FEA (plant specific) |
| W-ID: 4.2 Core Barrel Assembly – Middle Axial Welds | Limit load, EPFM, or LEFM analysis.<br>PWROG generic flaw evaluation complete. |
| W-ID: 4.3 Core Barrel Assembly – Lower Axial Welds | Limit load, EPFM, or LEFM analysis.<br>PWROG generic flaw evaluation complete. |
| W-ID: 4.4 Lower Support Assembly – Lower Support Column Bodies (Both Cast and Non-Cast) | Pattern analysis specific to lower support structure design and loads.<br>PWROG functionality analysis complete. |
| W-ID: 5 Baffle-former Assembly – Baffle-Edge Bolts | FMEA should address plant-specific practices and priorities. Some generic work possible to outline issues and options to be addressed in FMEA. |



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**Table 6-2 Applicability of Potential Generic Acceptance Criteria for Westinghouse-Design Primary and Expansion Component Items**

| Westinghouse Component | Approach |
|---|---|
| W-ID: 6 Baffle-former Assembly – Baffle-Former Bolts | Generic work completed in previous PWROG program.<br>Plant-specific analysis is generally necessary to develop acceptable bolting patterns. |
| W-ID: 6.1 Core Barrel Assembly – Barrel-Former Bolts | Generic work completed in previous PWROG program.<br>Plant-specific analysis is generally necessary to develop acceptable bolting patterns. |
| W-ID: 6.2 Lower Support Assembly – Lower Support Column Bolts | Generic program to share first-of-a-kind effort.<br>Pilot analysis of lower support structure to identify critical issues.<br>Expect final acceptance based on plant-specific consideration. |
| W-ID: 7 Baffle-former Assembly – Assembly | FMEA should address plant-specific practices and priorities. Some generic work possible to outline issues and options to be addressed in FMEA. |
| W-ID: 8 Alignment and Interfacing Components – Internals Hold Down Spring | Value determined by plant-specific design requirements. |
| W-ID: 9 Thermal Shield Assembly – Thermal Shield Flexures | Plant-specific analysis. Could be adapted from a generic dynamic model. |



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# APPENDIX A
# ACCEPTANCE CRITERIA METHODOLOGY AND DATA REQUIREMENTS FOR B&W PRIMARY AND EXPANSION COMPONENT ITEMS

## A.1 PRIMARY COMPONENT ITEMS

Acceptance criteria methodology and data requirements for each of the Primary component items are summarized in this appendix. A separate sub-section is provided for each component item using the following format:

- Component item identification and background information
- Component item function(s), including whether or not it has a core support safety function
- Observable effect(s)
- Possible examination outcome(s)
- Methodology for development of acceptance criteria
- Data requirements for development of acceptance criteria
- Existing documents (e.g., PWROG or Framatome)

Flowcharts corresponding with each component item are included in Appendix B. The flowcharts are step-by-step guides for completing an evaluation of each component item, whereas the text included in Appendix A forms the basis for each step.

Note that repairs or replacements, although not part of the scope of this document, are potential options for various components that are not summarized in this appendix (Appendix A), but are included in Appendix B.



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## A.1.1 Core Clamping Items

**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 potential exists for this to be a high wear location with subsequent reduced or loss of core clamping preload. This is a unique configuration with the B&W units and no known OE history indicating wear currently exists.

Component Item Function

The purpose of the core clamping load is to stabilize and significantly restrict the rigid body pendulum motion of the core support and plenum assemblies. In other words, the clamping action prevents rigid body rotation at the interface area. The clamping action does not have a direct core support safety function. Loss of clamping would undoubtedly lead to core barrel motions that would eventually lead to reactor shut down.

Observable Effects

A one-time physical measurement is to be obtained. The interference measurement to the nearest 0.001 inch is to be recorded at eight locations at approximately 45-degree intervals. Subsequent follow-up visual (VT-3) examinations are to be obtained.

ONS-1, ONS-2, ONS-3, ANO-1, TMI-1, DB-1, and CR-3 have performed physical wear measurements on a unit-specific basis and no relevant indications have been observed.

The physical measurement is performed to determine the differential height of the top of the plenum cover assembly weldment rib pads and plenum cover support ring to the reactor vessel seating surface. The measurement is the stack-up of the core support flange and the plenum support area versus the reactor vessel support ledge. The measurement must be taken without fuel in the reactor to eliminate the effect of the fuel hold-down springs. This interference/clearance fit was measured during original site assembly and also as-built measurements were taken of the piece parts after fabrication. The interference/clearance fit ranged between 0.004 to -0.0004 inch for the operating units.

Identification of an unacceptable condition is the precursor to perform follow-on investigations or analysis to establish the extent of degradation. The subsequent VT-3 inspections are to look for wear on the top and bottom surface of the CSS top flange, bottom surface of the plenum cover assembly support flange, and the top surface of the plenum cover assembly weldment rib pads and plenum cover support ring.

Possible examination outcomes

- No wear observed (data fall within scatter of original measurements)
- Wear of some extent is observed (one location, several locations, etc.)
- Other relevant conditions identified (e.g., missing rib pad)



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Methodology and Data Requirements

The original 0.004-inch criterion is based on the estimated permanent deformation in the O-ring mating surfaces, which is not considered in the interference measurement. The original analytical solution predicted adequate core clamping accounting for flow loadings with initial interference fit at zero also not accounting for the permanent deformation. The acceptance criterion is based on engineering judgment, and is defined as a reduction of no greater than 0.004 inch compared to the original as-built data. The original as-built results ranged from 0.004-inch interference to a 0.0004-inch clearance. Seven B&W units have now been measured with no appreciable change in interference from the original as-built results. The criterion of 0.004 inch reduction in interference is not to be construed as an indication of inadequate core clamping, but an indication that the surface conditions have changed since the unit was put into service. Additional inspection, using VT-3 for example, will be required to verify that wear is actually occurring.

The general analytical methodology to be used for determining the wear acceptance criterion involves the following steps and inputs:

- Determine the minimum core clamping preload required
    - This requires the differential pressure distribution on the core support cylinder due to reactor coolant flow
    - This also includes evaluation for different minimum preloads versus time at different operating conditions
- Determine an uncertainty margin, which includes both input pressure differential and measurement error
- Determine how the clamp load varies with operating conditions
- Develop a wear estimate for time from discovery of wear to time for required remediation

The VT-3 exam acceptance criteria are “no relevant conditions.” Relevant conditions for the B&W Items of interest are defined in Section 5 of MRP-227-1A. These relevant conditions constitute discontinuities or imperfections (e.g., loose or missing parts, distortion, corrosion, wear, etc.), which are readily detectable with the VT-3 technique. Note that the VT-3 exam method meets the requirements of ASME B&PV Code Section XI, as specified and conditioned by 10 CFR 50.55a, and the MRP-228 Inspection Standard.

Existing Documentation

- In the design stage, a design clamping load at operational conditions was established based on the flow uplift loading and horizontal forces developed by various pump combinations. An additional study in 1974 indicated that the clamping value was marginal using the reactor vessel stud preload identified in the reactor vessel instruction manual. A new preload was incorporated into the design, which provided satisfaction of the criteria at operating conditions but indicated there may be some loss of clamping at low temperatures (approximately < 300°F). It was verified by observation that the potential loss of preload was not leading to short-term degradation of the clamping surfaces.
- A generic core clamping evaluation has been performed
- No acceptable evaluation or analysis has been completed to date for determining a re-inspection schedule.
    - A wear rate estimate would be needed to project the wear over one cycle or more for continued operation



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## A.1.2 Control Rod Guide Tube Spacer Castings

**B2. Control Rod Guide Tube Assembly**

CRGT spacer castings

CRGT spacer castings are subject to thermal aging embrittlement, which if a flaw would be present and they are subjected to loading that exceeds the materials degraded fracture toughness, such a condition could potentially lead to cracking. There is no known cracking of CRGT spacer castings.

Component Item Function

The function of the spacer castings is to provide structural support to the 12 perforated vertical rod guide tubes and 4 pairs of vertical rod guide sectors within each CRGT assembly. Degradation of the spacer castings could result in degradation in the unit shutdown capability. The spacer castings do not have a core support safety function; however, they do have a safety function relative to control rod alignment, insertion and reactivity issues. Degradation of the spacer castings could result in degradation in the unit shutdown capability by hindering the insertion of the control rods into the core in the normal anticipated time.

Observable Effects

A visual (VT-3) examination of the CRGT spacer castings is to be performed. Subsequent visual (VT-3) examinations are to be obtained.

The spacer castings have limited accessibility from the top or bottom of the CRGT through a center free-path (once the plenum assembly is removed from the vessel). Examination at the quarter points where the threaded connections are present is possible. These lanes are not blocked by the rod guide tubes. The examination would look for cracking of the spacer surface or evidence that the spacer is not approximately centered. The threaded fasteners are welded to the OD of the pipe column so it is possible that a degraded threaded location would not be detected.

Possible examination outcomes

- No relevant conditions identified
- One or more areas are identified with a large crack-like indication that may be a precursor to losing a piece of material
- One or more areas are identified with missing material, including one or more missing screws
- One or more anomalies near the threaded region
- Evidence that the spacer casting is not centered

Methodology and Data Requirements

The general analytical methodology to be used for acceptance criteria for the CRGT spacer castings involves the following steps and input:

- Perform a degraded casting evaluation to determine the extent of degradation allowable such that unacceptable drop times of the control rods will not occur or the control rods will be unable to pass through the CRGT assembly

In the event that the above effort is unsuccessful, the following action is to be taken:



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- Perform a reactivity analysis to determine the number of CRDMs that are required for shut down of the reactor

The VT-3 exam acceptance criteria are “no relevant conditions.” Relevant conditions for the B&W Items of interest are defined in Section 5 of MRP-227-1A. These relevant conditions constitute discontinuities or imperfections (e.g., loose or missing parts, distortion, corrosion, wear, etc.), which are readily detectable with the VT-3 technique. Note that the VT-3 exam method meets the requirements of ASME B&PV Code Section XI, as specified and conditioned by 10 CFR 50.55a, and the MRP-228 Inspection Standard.

The licensee shall submit the plant-specific analysis for this component item within one year after the inspection if the results do not meet the acceptance criteria in MRP-227.

Existing Documentation

- Current licensing basis (CLB) loadings (normal and faulted condition) are not currently available or even possibly required for analyses, but could be easily developed
- Core design
- Unit-specific and generic engineering evaluations and assessments (aka functionality analyses) have been performed



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## A.1.3 Vent Valve Assembly Vent Valve Retaining Rings

**B3. Vent Valve Assembly**

(a) Vent valve top retaining ring

(b) Vent valve bottom retaining ring

Vent valve top and bottom retaining rings are subject to thermal aging embrittlement, which if a flaw would be present and they are subjected to loading that exceeds the materials degraded fracture toughness, such a condition could potentially lead to cracking. Although there have been instances of failures of precipitation-hardenable materials in other applications, there is no known OE identifying cracking in PWR reactor vessel internals applications.

Component Item Function

Vent valve assemblies are generally passive devices (although the hinge assembly and disc are identified as active components) with the primary function of relieving pressure in the interior of the core support assembly during a cold leg large break loss-of-coolant-accident (LOCA). For all normal operating conditions the vent valves are closed. Another function of the vent valve is to prevent unacceptable bypass flow during normal operation. The vent valve top and bottom retaining rings do not have a core support function; however, failure of the vent valve top and bottom retaining rings could cause the vent valve body to be pushed inward, becoming lodged in the annulus between the plenum and the core support shield, which would increase the core bypass flow and also cause loose parts to be released into the reactor coolant system.

Observable Effects

A visual (VT-3) examination of the accessible surfaces of the vent valve top and bottom retaining rings is to be performed. Subsequent visual (VT-3) examinations are to be obtained.

The vent valve top and bottom retaining rings are being examined to detect damage either in the form of a precursor to the loss of material, or, a piece or section of the material that has fractured or is currently missing, particularly, in the areas where high stresses exist. For example, for the retaining rings, at the locations where the threaded jacking screws are connected.

Possible examination outcomes

- No relevant conditions identified
- Observation that either retaining ring is not in the correct position
- One or more areas are identified with crack-like indications
- One or more areas are identified with missing material
- Evidence one or more threaded connection has failed

Methodology and Data Requirements

The general analytical methodology to be used for acceptance criteria for the vent valve top and bottom retaining rings involves the following steps and inputs:

- Perform analyses or evaluations, which will determine areas of the rings, if any, where the thermal embrittlement effects may be expected to result in cracking of these component items



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- Summarize existing vent valve opening and flow assessments
- Perform an assessment of operational loading relative to critical crack size in the degraded condition
- A VT-1, ET, or UT examination may be needed to determine if flaws are emanating from the location where missing material may be identified

The VT-3 exam acceptance criteria are "no relevant conditions." Relevant conditions for the B&W Items of interest are defined in Section 5 of MRP-227-1A. These relevant conditions constitute discontinuities or imperfections (e.g., loose or missing parts, distortion, corrosion, wear, etc.), which are readily detectable with the VT-3 technique. Note that the VT-3 exam method meets the requirements of ASME B&PV Code Section XI, as specified and conditioned by 10 CFR 50.55a, and the MRP-228 Inspection Standard.

The licensee shall submit the plant-specific analysis for this component item within one year after the inspection if the results do not meet the acceptance criteria in MRP-227.

Existing Documentation

- CLB loadings (normal and faulted condition) are available
- Procurement and fabrication data need to be identified to identify the non-destructive examinations performed on the as-installed retaining rings
- No acceptable evaluation or analysis has been completed to date for determining a re-inspection schedule
- Unit-specific and generic stress analyses have been performed



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## A.1.4 Vent Valve Assembly Original Locking Devices

### B4, B5. Vent Valve Assembly

Original locking devices (pressure plate, spring retainer, spring, U-cover)

Original locking devices (key ring, pin)

Parts in the original locking devices (pressure plate, spring retainer, spring, U-cover) are potentially susceptible to wear. Other parts in the original locking devices (key ring, pin) are potentially susceptible to thermal embrittlement, which in the presence of a flaw could result in cracking. While these degradation mechanisms could eventually result in an impact on the function of the vent valve assembly, an impact on the function of the vent valves has not been observed to date.

Component Item Function

Vent valve assemblies are installed in the mounting rings in the core support shield. For all normal operating conditions, the vent valve is closed. In the event of a cold leg large break LOCA, the valve will open to vent steam generated in the core directly to the break, thus permitting the core to be flooded and adequately cooled after emergency core coolant has been supplied to the reactor vessel. Each vent valve assembly consists of a hinged disc, a valve body with sealing surfaces, two retaining rings and fasteners, and an alignment device (to maintain the correct orientation). Each vent valve assembly can be remotely handled as a unit for removal or installation. Vent valve component parts, including the disc, are designed to minimize the possibility of loss of parts to the reactor coolant system, and all operating fasteners include a positive locking device. The hinged-disc includes a device for remote testing and verification of proper disc function. The external side of the disc is contoured to absorb the impact load of the disc on the RV inside wall without transmitting excessive impact loads to the hinge parts as a result of a LOCA.

The jackscrew locking devices prevent the jackscrews from turning during service.

Observable Effects

A visual (VT-3) examination of 100% the accessible surfaces, including the overall valve symmetry in the mounting ring and the overall jackscrew thread extension from the lower retaining ring threaded flange, is to be performed. Subsequent visual (VT-3) examinations are to be performed.

The vent valve locking devices are being examined to detect loss of material from the original locking device (wear associated with jack screw spring and pressure plate) and cracking of the original locking devices (key ring and pin).

The specific relevant conditions for the original locking devices are: 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, observations that the jackscrew is out of the design configuration (e.g., observation of a broken or cracked jackscrew, incorrect engagement of the jackscrews), and observations that the valve is not symmetrical in the mounting ring or the jackscrew thread extensions from the lower retaining ring threaded flange are unequal.

Possible examination outcomes

- No relevant and/or reportable conditions identified



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- Observation that the vent valve is not symmetric in the mounting ring
- Observation that the overall jackscrew thread extension from the lower retaining ring is unequal and/or incorrect
- Observation the jackscrew is out of design configuration
- Observation of U-cover damage, wear, and/or incorrect positioning
- Observation of pressure plate damage and/or misalignment

Methodology and Data Requirements

The general analytical methodology to be used for acceptance criteria for the vent valve locking devices involves the following steps and inputs:

- Determine impact of observation(s) on future function of component item(s), weld(s), and overall vent valve function and/or design configuration
- Determine if the vent valve assembly can be left as-is (with observations) or if any actions (up to vent valve replacement) need to be taken for the vent valve (both component item and overall vent valve) to perform its intended function

Existing Documentation

- Several previous analyses of observations have been performed with varying results (vent valve left as-is versus vent valve replacement)
- Bypass leakage calculations are available
- Exercise load calculations are available
- Original design calculations, such as vent valve opening force calculations, are available



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## A.1.5 Vent Valve Assembly Modified Locking Devices

**B6. Modified locking devices**

Modified locking devices (bolt locking cup, jackscrew locking cup, bolted block)

Parts in the modified locking devices (bolt locking cup, jackscrew locking cup, bolted block) are potentially susceptible to SCC. While this could eventually result in an impact on the function of the vent valve assembly, an impact on function of the vent valves has not been observed to date.

Component Item Function

Vent valve assemblies are installed in the mounting rings in the core support shield. For all normal operating conditions, the vent valve is closed. In the event of a cold leg large break LOCA, the valve will open to vent steam generated in the core directly to the break, thus permitting the core to be flooded and adequately cooled after emergency core coolant has been supplied to the reactor vessel. Each vent valve assembly consists of a hinged disc, a valve body with sealing surfaces, two retaining rings and fasteners, and an alignment device (to maintain the correct orientation). Each vent valve assembly can be remotely handled as a unit for removal or installation. Vent valve component parts, including the disc, are designed to minimize the possibility of loss of parts to the reactor coolant system, and all operating fasteners include a positive locking device. The hinged-disc includes a device for remote testing and verification of proper disc function. The external side of the disc is contoured to absorb the impact load of the disc on the RV inside wall without transmitting excessive impact loads to the hinge parts as a result of a LOCA.

The jackscrew locking devices prevent the jackscrews from turning during service.

Observable Effects:

A visual (VT-3) examination of 100% the accessible surfaces, including the overall valve symmetry in the mounting ring and the overall jackscrew thread extension from the lower retaining ring threaded flange, is to be performed. Subsequent visual (VT-3) examinations are to be performed.

The modified vent valve locking devices are being examined to detect cracking, including detection of fractured or missing locking cups and welds, or the bolted block.

The specific relevant conditions for the modified locking devices are: 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, a fractured or missing bolted block, missing cap screws associated with the crimped locking cups, observations that the jackscrew thread extensions from the lower retaining ring threaded flange are unequal, and observations that the valve is not symmetrical in the mounting ring.

Possible examination outcomes

- No relevant and/or reportable conditions identified
- Observation that the vent valve is not symmetric in the mounting ring
- Observation that the overall jackscrew thread extension from the lower retaining ring is unequal and/or incorrect
- Observation the jackscrew is out of design configuration
- Observation of a cracked or fractured locking cup and/or locking cup weld



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- Observation of a missing locking cup and/or locking cup weld
- Observation of a cracked or fractured missing bolted block
- Observation of a missing bolted block
- Observation of a torn crimp or visual evidence that the crimp and sleeve slot are not aligned or engaged
- Observation of missing cap screws associated with the locking cups

Methodology and Data Requirements

The general analytical methodology to be used for acceptance criteria for the vent valve locking devices involves the following steps and inputs:

- Determine impact of observation(s) on future function of component item(s), weld(s), and overall vent valve function and/or design configuration
- Determine if the vent valve assembly can be left as-is (with observations) or if any actions (up to vent valve replacement) need to be taken for the vent valve (both component item and overall vent valve) to perform its intended function

Existing Documentation

- Several previous analyses of observations have been performed with varying results (vent valve left as-is versus vent valve replacement)
- Bypass leakage calculations are available
- Exercise load calculations are available
- Original design calculations, such as vent valve opening force calculations, are available



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## A.1.6 Upper Core Barrel Bolts and Locking Devices

### B7. Core Support Shield Assembly

Upper core barrel (UCB) bolts and their locking devices

There is a potential for intergranular stress corrosion cracking (IGSCC) of Alloy A-286 and Alloy X-750 bolting. Past B&W failure history exists with the original Alloy A-286 bolt materials in B&W-design units and with applications of Alloy X-750 material within the nuclear industry (in general). Currently, there are no known UT indications of any of the replacement bolts (Alloy A-286 or Alloy X-750) in the operating B&W-design units. There are two instances of a UT indication or suspect indication where an original Alloy X-750 bolt is still in service in the operating B&W fleet (installed at TMI-1 only).

Component Item Function

The UCB joint carries the entire weight of the core and the majority of the weight of the reactor vessel internals. The UCB bolts perform a core support function since joint failure would result in a drop of the lower core support assembly, including the core itself. The guide lugs welded to the inside wall of the reactor vessel are designed to limit this drop so that the ability to shut down the plant is maintained.

Observable Effects

A volumetric examination (UT) of the bolts and a visual (VT-3) examination of the bolt locking devices are to be performed. Subsequent visual (VT-3) examinations of the locking devices and ultrasonic (UT) examinations of the bolts are to be performed.

Cracking of the bolts is the main concern and the locking devices are to be examined to identify if any are damaged, distorted, loose, broken, or missing.

PWROG work evaluated the potential information that could be determined from only a visual examination of the bolt and locking devices.

Possible examination outcomes

*Bolts*

Any cracking is anticipated to be located at the head-to-shank area where the peak tensile stress exists (i.e., a SCF exists) and OE has shown high-strength bolting to crack. It is also possible that cracking may occur in the shank thread region where high tensile stress may exist.

- No relevant conditions identified
- Relevant conditions (i.e., crack-like indications, either completely cracked or partially cracked; or non-interpretable UT indications, such as no back wall reflection or multiple reflections with no crack-like indication that is most likely caused by a large or duplex grain size) are identified
    - One or a few bolts (exact number is unit-specific) are identified with relevant indications
    - More than a few bolts (exact number is unit-specific) are identified with relevant indications

*Locking Devices*

- No relevant conditions identified



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- One or two are identified with damage, distortion, or are missing
- More than two bolt locking devices are identified with damage, distortion, or are missing

Methodology and Data Requirements

*Bolts*

The general analytical methodology to be used for acceptance criteria for the bolts (i.e., acceptable bolt pattern) and to justify an inspection interval involves the following steps and inputs if relevant conditions have been identified in the UCB bolts:

Methodology for an acceptable bolt pattern analysis (ABPA)

- A finite element model (FEM) is to be developed for the local geometry with contact conditions, pretension elements, loads and boundary conditions
- A thermal analysis is to be performed
    - Determines bolt temperatures and temperature gradients for normal operating conditions
- A structural analysis is to be performed in which failed bolts are inactive
    - Stress concentration factors are calculated to determine the peak stresses at the bolt head-to-shank fillet region under normal operating conditions
    - Analysis is performed for all loads and load combinations required for an ASME Code evaluation (stress limits for threaded structural fasteners in subsection NG and Appendix F)
    - The effect of failed or missing bolts on overall effective core barrel stiffness is evaluated
        - A change of no more than 20% in stiffness when subjected to LOCA loads is acceptable (within the limits of other uncertainties accounted for in the evaluation of LOCA loadings)
            - This corresponds to approximately 10% change in fundamental frequency of the structure
    - An evaluation of joint stability (i.e., openness) is also to be performed
- Representative rejected UCB bolts may be removed for laboratory testing to confirm the UT inspection results and IGSCC mechanism, or to identify any other failure mechanism(s)

NOTE: One alternative to removing representative UT rejected bolts for laboratory examination is to re-inspect all bolts at the next refueling outage if continued operation for one additional fuel cycle can be supported by a technical evaluation. Other possible options such as replacement of a predefined bolt pattern may also be pursued.



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Methodology for using the ABPA to justify a re-inspection interval

- Based on the results of UCB bolt UT inspection and any laboratory test results, perform an evaluation to predict the number of failed bolts at the end of the re-inspection interval. The evaluation will use a bolt degradation model based on OE and/or mechanistic models of the degradation mechanism(s). An empirical and/or empirical and mechanistic model of the bolting degradation effect is developed from industry OE, plant-specific design information and/or test data. The model is calibrated to plant-specific bolt inspection results; or otherwise applies a bolt degradation model for similar plants after accounting for relevant differences, as appropriate. The changes to the peak stress at the bolt head-to-shank fillet region as a result of the identified failures should be included for evaluation of increased susceptibility to SCC.
- Incorporate the effect of future UCB bolt failure into the operability evaluation and re-inspection requirement. The predicted number of failed bolts at the end of the re-inspection interval determined in the previous step will be used to project a statistically significant number of future bolting patterns (e.g. 10,000 samples) that may exist at the end of the re-inspection interval.
- The projected statistically significant number of future bolting patterns (e.g. 10,000 samples) that may exist at the end of the re-inspection interval will be compared to the results of the ABPA to justify at least a 95% probability of an acceptable bolt pattern at the end of the re-inspection interval. It is anticipated that this method will be used to demonstrate a 10-year re-inspection interval. Re-inspection intervals beyond 10-years may be justifiable in certain circumstances.

*Locking Devices*

The VT-3 exam acceptance criteria are “no relevant conditions.” Relevant conditions for the B&W Items of interest are defined in Section 5 of MRP-227-1A. These relevant conditions constitute discontinuities or imperfections (e.g., loose or missing parts, distortion, corrosion, wear, etc.), which are readily detectable with the VT-3 technique. Note that the VT-3 exam method meets the requirements of ASME B&PV Code Section XI, as specified and conditioned by 10 CFR 50.55a, and the MRP-228 Inspection Standard.

Data Requirements

Due to the variations in bolt materials used and loadings among the units, unit-specific analytical efforts are required. The generic efforts have already been completed in PWROG work.

Existing Documentation

- A generic UCB bolt stress analysis was completed in the MRP reactor internals project for the three ONS units and ANO-1, which was subsequently used in the PWROG work noted below
- FEM and thermal analysis have been developed by the PWROG project to evaluate failures for use in evaluating an acceptable failure pattern or number of failed UCB and LCB bolts allowed for continued operation, and for use in preparing a justification for continued operation (JCO)
    - One or a few bolts could be identified as failed (non-interpretable bolt UT signal; inaccessible bolt for UT; locking device observed to be missing, non-functional, or removed; partially cracked bolt; or completely cracked bolt) and shown to be acceptable with no further action needed
    - A number of bolts (TBD) identified as failed (non-interpretable UT signal, partially cracked, or completely cracked) would initiate replacement activities and lead into the expansion category
- Some unit-specific analyses for several assumed inspection cases have been performed for each unit in the PWROG project, but unit-specific analyses may be re-performed for the actual inspection result



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- If inspection results indicate no relevant indications of failure and calculated peak stresses are below the bolt material yield strength, SCC is not expected to initiate and an inspection during the next 10-year ISI interval is judged adequate



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## A.1.7 Lower Core Barrel Bolts and Locking Devices

**B8. Core Barrel Assembly**

Lower core barrel (LCB) bolts and their locking devices

There is a potential for intergranular stress corrosion cracking (IGSCC) of Alloy A-286 and Alloy X-750 bolting. Past B&W failure history exists with the original Alloy A-286 bolt materials in B&W-design units and with applications of Alloy X-750 material within the nuclear industry (in general). Currently, there are no known UT indications of any of the replacement bolts (Alloy A-286 or Alloy X-750) in the operating B&W-design units. There are two instances where an original Alloy X-750 bolt is still in service in the operating B&W fleet (installed at TMI-1 only).

Component Item Function

The LCB joint carries the entire weight of the core and the weight of the lower grid of the reactor vessel internals. The LCB bolts perform a core support function since joint failure would result in a drop of the core and lower grid. The guide lugs welded to the inside wall of the reactor vessel are designed to limit this drop such that the ability to shut down the plant is maintained.

Observable Effects

A volumetric examination (UT) of the bolts and a visual (VT-3) examination of the bolt locking devices are to be performed. Subsequent visual (VT-3) examinations of the locking devices and ultrasonic (UT) examinations of the bolts are to be performed.

Cracking of the bolts is the main concern and the locking devices are to be examined to identify if any are damaged, distorted, loose, broken, or missing.

PWROG work evaluated the potential information that could be determined from only a visual examination of the bolt and locking devices.

Possible examination outcomes

*Bolts*

Any cracking is anticipated to be located at the head-to-shank area where the peak tensile stress exists (i.e., a SCF exists) and OE has shown high-strength bolting to crack. It is also possible that cracking may occur in the shank thread region where high tensile stress may exist.

- No relevant conditions identified
- Relevant conditions (i.e., crack-like indications, either completely cracked or partially cracked; or non-interpretable UT indications, such as no back wall reflection or multiple reflections with no crack-like indication that is most likely caused by a large or duplex grain size) are identified
    - One or a few bolts (exact number is unit-specific) are identified with relevant indications
    - More than a few bolts (exact number is unit-specific) are identified with relevant indications

*Locking Devices*

- No relevant conditions identified



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- One or two are identified with damage, distortion, or are missing
- More than two bolt locking devices are identified with damage, distortion, or are missing

Methodology and Data Requirements

*Bolts*

The general analytical methodology to be used for acceptance criteria for the bolts (i.e., acceptable bolt pattern) and to justify an inspection interval involves the following steps and inputs if relevant conditions have been identified in the LCB bolts:

Methodology for ABPA

The general analytical methodology to be used for acceptance criteria for the bolts involves the following steps and inputs if relevant conditions have been identified in the LCB bolts:

- A finite element model (FEM) is to be developed for the local geometry with contact conditions, pretension elements, loads and boundary conditions
- A thermal analysis is to be performed
    - Determines bolt temperatures and temperature gradients for normal operating conditions
- A structural analysis is to be performed in which failed bolts are inactive
    - Stress concentration factors are calculated to determine the peak stresses at the bolt head-to-shank fillet region under normal operating conditions
    - Analysis is performed for all loads and load combinations required for an ASME Code evaluation (stress limits for threaded structural fasteners in subsection NG and Appendix F)
    - The effect of failed or missing bolts on overall effective core barrel stiffness is evaluated
        - A change of no more than 20% in stiffness when subjected to LOCA loads is acceptable (within the limits of other uncertainties accounted for in the evaluation of LOCA loadings)
            - This corresponds to approximately 10% change in fundamental frequency of the structure
    - An evaluation of joint stability (or, openness) is also to be performed
- Representative rejected LCB bolts may be removed for laboratory testing to confirm the UT inspection results and IGSCC mechanism, or to identify any other failure mechanism(s)

NOTE: One alternative to removing representative UT-rejected bolts for laboratory examination is to re-examine all bolts at the next refueling outage if continued operation for one additional fuel cycle can be supported by a technical evaluation. Other possible options such as replacement of a predefined bolt pattern may also be pursued.



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Methodology for using the ABPA to justify a re-inspection interval

- Based on the results of LCB bolt UT inspection and any laboratory test results, perform an evaluation to predict the number of failed bolts at the end of the re-inspection interval. The evaluation will use a bolt degradation model based on OE and/or mechanistic models of the degradation mechanism(s). An empirical and/or empirical and mechanistic model of the bolting degradation effect is developed from industry OE, plant-specific design information and/or test data. The model is calibrated to plant-specific bolt inspection results; or otherwise applies a bolt degradation model for similar plants after accounting for relevant differences, as appropriate. The changes to the peak stress at the bolt head-to-shank fillet region as a result of the identified failures should be included for evaluation of increased susceptibility to SCC.
- Incorporate the effect of future LCB bolt failure into the operability evaluation and re-inspection requirement. The predicted number of failed bolts at the end of the re-inspection interval determined in the previous step will be used to project a statistically significant number of future bolting patterns (e.g. 10,000 samples) that may exist at the end of the re-inspection interval.
- The projected statistically significant number of future bolting patterns (e.g. 10,000 samples) that may exist at the end of the re-inspection interval will be compared to the results of the ABPA to justify at least a 95% probability of an acceptable bolt pattern at the end of the re-inspection interval. It is anticipated that this method will be used to demonstrate a 10-year re-inspection interval. Re-inspection intervals beyond 10-years may be justifiable in certain circumstances.

*Locking Devices*

The VT-3 exam acceptance criteria are "no relevant conditions." Relevant conditions for the B&W Items of interest are defined in Section 5 of MRP-227-1A. These relevant conditions constitute discontinuities or imperfections (e.g. loose or missing parts, distortion, corrosion, wear, etc.), which are readily detectable with the VT-3 technique. Note that the VT-3 exam method meets the requirements of ASME B&PV Code Section XI, as specified and conditioned by 10 CFR 50.55a, and the MRP-228 Inspection Standard.

Data Requirements

Due to the variations in bolt materials used and loadings among the units, unit-specific analytical efforts are required. The generic efforts have already been completed in PWROG work.

Existing Documentation

- A generic LCB bolt stress analysis was completed in the MRP reactor internals project for the three ONS units and ANO-1, which was subsequently used in the PWROG work noted below.
- FEM and thermal analysis have been developed by the PWROG project to evaluate failures for use in evaluating an acceptable failure pattern or number of failed UCB and LCB bolts allowed for continued operation, and for use in preparing a JCO
    - One or a few bolts could be identified as failed (non-interpretable bolt UT signal; inaccessible bolt for UT; locking device observed to be missing, non-functional, or removed; partially cracked bolt; or completely cracked bolt) and shown to be acceptable with no further action needed
    - A number of bolts (TBD) identified as failed (non-interpretable UT signal, partially cracked, or completely cracked) would initiate replacement activities and lead into the expansion category
- Some unit-specific analyses for several assumed inspection cases have been performed for each unit in the PWROG project, but unit-specific analyses may be re-performed for the actual inspection result.



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- If inspection results indicate no relevant indications of failure and calculated peak stresses are below the bolt material yield strength, SCC is not expected to initiate and an inspection during the next 10-year ISI interval is judged adequate.
- Unit-specific analyses have been performed to assess future LCB bolt failure potential and justify a re-examination interval



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## A.1.8 Baffle-to-Former Bolts

**B9. Core Barrel Assembly**

Baffle-to-former bolts

There is a potential for failure in the form of cracking of baffle-to-former bolting to occur mainly from irradiation-assisted stress corrosion cracking (IASCC), but also as a result of irradiation embrittlement, irradiation creep/stress relaxation (leading to fatigue and wear), or overload (from a prying effect). Past failure history exists with baffle-to-former bolt materials (Type 316CW and Type 347) and core barrel-to-former bolt materials (Alloy X-750 and Type 316Ti CW) in non-B&W-design units. Currently, there is one known reportable UT indication of failure of a Type 304 baffle-to-former bolt in the operating B&W-design units.

Component Item Function

The core barrel assembly consists of the core barrel cylinder, former plates, and baffle plates connected by bolted joints that include: (1) core barrel-to-former bolts (CF bolts), (2) baffle-to-former bolts (BF bolts), and (3) baffle-to-baffle bolts (BB bolts). The core barrel assembly supports the fuel assemblies, lower grid, flow distributor, and incore instrument guide tubes. The baffle plates, former plates, and their joints (including BF bolts) do not have a core support function and are categorized as internals structures. The primary function of the baffle plates, former plates, and their bolted connections is to provide a flow envelope surrounding the core. Also, since they are bolted to the core barrel cylinder, the baffle plates and former plates will produce a small increase on the stiffness and natural frequencies of the core barrel assembly.

The CF and BF bolts have the function of maintaining structural integrity of the baffle and former portion of the structural assembly and thus of maintaining flow geometry during normal operation. For faulted events, a small number of the CF and BF bolts are needed to restrain the baffle so that a coolable core geometry is maintained.

Observable Effects

Cracking of the bolts is the main concern. A volumetric examination (UT) of the bolts is to be performed to identify such cracking.

Possible examination outcomes

Cracking is anticipated to occur at the head-to-shank area where the peak tensile stress exists (i.e., a SCF exists) and OE has shown them to crack at this location in the past, although it may also occur in the shank thread region where high tensile stress is also possible.

- No relevant conditions identified
- Relevant conditions (i.e., crack-like indications, either completely cracked or partially cracked; or non-interpretable UT indications, such as no back wall reflection or multiple reflections with no crack-like indication that is most likely caused by a large or duplex grain size) are identified

Methodology and Data Requirements

Goal: Must demonstrate that bolting patterns meet acceptance criteria for continued operation through the re-inspection interval



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Data Requirements:

- Loads
- UT inspection results
- Updated industry OE and test data for bolt failure empirical model (as necessary)
- Baffle-former assembly design information including any applicable modifications
- Fast neutron (dpa) distribution in the baffle-barrel region (if needed)

Overall Approach

Structural integrity of the baffle-former assembly must be demonstrated with an ABPA, which has not yet been established for B&W units. The ABPA is created using a finite element based deterministic analysis of an acceptable bolt pattern and engineering evaluation to demonstrate that the bolt pattern is in compliance with the design basis of the unit, or class of unit.

An ABPA shall contain the following elements:

1. Development of pressure and displacement boundary conditions on the baffle-former-barrel assembly and upper and lower grids for faulted events (LOCA, SSE), in accordance with unit licensing basis.
2. Demonstration that bolt stress calculated with a linear-elastic material model remains below faulted allowable limits for linear-elastic material. For replacement bolts, the faulted allowable limits are that $P_m$ shall not exceed the lesser of $2.4*S_m$ and $0.7*S_u$ and that $P_m + P_b$ shall not exceed 150% of the $P_m$ limit, where $S_m$ and $S_u$ are based on ASME Code material properties. For irradiated bolts, the faulted allowable limits are that $P_m < (2/3)*Su$ and $P_m + P_b \leq S_u$, where $S_u$ is the 95$^{th}$ percentile lower bound irradiated ultimate strength of the bolt.
3. Evaluation of BF bolts high cycle fatigue, BF bolts low cycle fatigue, core bypass flow, and baffle plate jetting.
4. Justification that core coolability and control rod insertability are maintained using approved methods from the unit specific licensing basis. This may be demonstrated by showing calculated fuel impact loads are less than the allowable limits.

Probabilistic Structural Analysis Methodology

Two methodologies are provided below: 1) for units with an acceptable bolting pattern analysis (ABPA) and 2) for units without an ABPA.

Units with an ABPA

For units with an ABPA, a probabilistic structural analysis of the baffle assembly will be used to demonstrate with 95% confidence that the reactor internals will meet the unit design bases at the end of the re-inspection interval. It is anticipated that this method will be used to demonstrate a 10-year re-inspection interval. Re-inspection intervals beyond 10-years or partial UT exam coverage may be justifiable in certain circumstances (e.g., future bolt pattern relies on replacement bolts or the plant is in, or has been converted to, an upflow configuration).

This methodology is described below:



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1. Develop and maintain a bolt degradation model based on OE and/or mechanistic models of the degradation mechanism(s). An empirical and/or empirical and mechanistic models of the bolting degradation effect is developed from industry OE, unit-specific design information and/or test data. The model is calibrated to unit-specific bolt inspection results or otherwise applies a bolt degradation model for similar units after accounting for relevant differences, as appropriate.
2. Qualify an efficient surrogate for finite element based ABPA that also demonstrates compliance with design basis.
3. Perform a probabilistic structural analysis to demonstrate that the future bolt pattern will be an Acceptable Bolt Pattern (ABP) at the end of the re-inspection interval.
    a. Starting with the latest inspection results, project future bolting patterns versus time, using the degradation model from Step 1 above.
    b. The projected future patterns shall consider the associated probability of detection of the inspection technique. Non-inspected bolts may be assumed as failed, or they may be credited when justified by analysis (e.g., statistical sampling of like populations from equivalent exams).
    c. When predicting the total quantity of failed bolts in future patterns, it is acceptable to consider results associated with the 95th fractile.
    d. The projected future patterns may credit bolt replacements or other unit modifications, as appropriate.
    e. Using the efficient surrogate from Step 2, demonstrate that at least 95% of projected future patterns are ABPs at the time of the next UT examination.
    f. A sufficient number of projected future patterns shall be evaluated to ensure convergence above the 95% threshold defined above.

The probabilistic structural analysis shall address the following items:

- Required plant-specific or bounding inputs
    - Basis for uncertainties of each input
    - Treatment of input uncertainty in degradation model and probabilistic structural analysis
- Description of degradation model and underlying principals
    - Summary of benchmarking studies demonstrating the validity of the degradation model and characterizing uncertainty in degradation model results
    - Treatment of degradation model uncertainty in probabilistic structural analysis
- Description of efficient surrogate structural analysis model and characterization of bias/uncertainty relative to finite element based ABPA
- Description of numerical convergence in probabilistic structural analysis demonstrating acceptability of 95% of predicted patterns

Units without an ABPA

While it is recommended that units have an ABPA, a one-cycle justification can be created in lieu of an ABPA when inspection results in a small number of failed BF bolts that are well-spaced. The following methodology may be used:



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1. Develop and maintain a bolt degradation model based on OE and/or mechanistic models of the degradation mechanism(s). An empirical and/or empirical and mechanistic models of the bolting degradation effect is developed from industry OE, unit-specific design information and/or test data. The model is calibrated to unit-specific bolt inspection results; or otherwise applies a bolt degradation model for similar units after accounting for relevant differences, as appropriate. This bolt degradation model will be used to predict the number and location of failed BF bolts at the end of the re-inspection interval.
2. The predicted number of failed bolts at the end of the re-inspection interval determined in the previous step will be used to project a statistically significant number of future bolting patterns (e.g., 10,000 samples) that may exist at the end of the re-inspection interval.
3. The projected future bolting patterns that may exist at the end of the re-inspection interval will be compared to bolt spacing rules to justify at least a 95% probability of maintaining the bolt spacing rules at the end of the re-inspection interval. The bolt spacing rules will be selected based on the fundamental understanding of how bolt failures may affect the limiting factors regarding maintaining design bases, which are bolt stress allowables, baffle-to-fuel impact loads, core bypass flow, and high cycle fatigue.
4. Using the results of the previous steps, a qualitative assessment will be performed to justify that the design bases will be maintained through the re-inspection interval. Note that, given the limitations of this method, it can only be applied when inspection results in a small number of failed BF bolts that are well-spaced.

Ability to Perform Generic Efforts

Analytical efforts for an ABPA could be performed on a generic basis for the B&W-design units, although there are two designs (ONS-1 and TMI-1 are one design and the other units are the second design). Analytical efforts for Probabilistic Structural Analysis can be performed on a generic basis, with unit specific input needed for inspection results and unit modifications and unit-specific or “class-of-unit” input needed for the ABPA. The NDE inspection standard for a class of unit could also be developed generically.

Existing Documentation:

- A FEM has been developed by the MRP Reactor Internals project that can be used in performing certain evaluations
    - A few bolts could be identified as failed (non-interpretable bolt UT signal; inaccessible bolt for UT; locking device observed to be missing, non-functional, or removed; partially cracked bolt; or completely cracked bolt) and shown to be acceptable with no further action needed
- A number of bolts (TBD) identified as failed (partially cracked or completely cracked, or bolts with a non-interpretable UT signal that cannot otherwise be statistically justified using appropriate sample sizes from representative bolts) would initiate replacement activities and lead into the expansion category
    - Past B&WOG work has determined the minimum number of bolts required for safe shutdown, but not for operation; i.e., no minimum bolt patterns have been determined
    - The prior B&WOG effort was not explicitly used in establishing expansion criteria; rather a combination of results obtained in MRP-229 and industry data detailed in MRP-231 were used



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- Bolt degradation model has been developed but would need to be updated based on most recent OE
- No acceptable evaluation or analysis has been completed to date for determining a re-inspection schedule



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## A.1.9 Baffle Plates

**B10. Core Barrel Assembly**

Baffle plates

Baffle plates are subject to irradiation embrittlement, so if a flaw is present and loading exists that exceeds the materials degraded fracture toughness, cracking could result. There is no known history of cracking of Type 304 baffle plate material in PWR reactor vessel internals applications. A recent examination at ONS-3 found a linear indication in the lower grid rib section (Type 304 material), but the exact nature of the indication has yet to be confirmed. To date, subsequent exams have reported no extension of the linear indication.

Component Item Function

Degradation of the baffle plates could result in increased core bypass flow and a reduction in margin to DNB, but would probably have a negligible effect on unit operations and would not be observed except by direct examination. The core barrel assembly supports the fuel assemblies, lower grid, flow distributor, and incore instrument guide tubes. However, the baffle plates do not support any load. The primary function of the baffle plates during normal power operation is to provide a flow envelope for the core and, thereby limit core bypass flow.

The baffle plates therefore do not have a direct core support function; however, they do have a safety function to control bypass around the core during a loss-of-coolant-accident (LOCA).

Observable Effects

A visual (VT-3) examination of the baffle plates is to be performed. Subsequent visual (VT-3) examinations are to be performed.

If flaws are identified, follow-on examination by VT-1, ET, or UT may be performed to characterize the length or both length and depth of observations.

The baffle plates are being examined to detect large surface cracks. The locations expected to be subjected to the highest tensile stresses are near the baffle bolt holes (baffle-to-former and baffle-to-baffle) and flow holes/slots. Examinations should also include areas near the HAZ of the baffle bolt locking devices where residual tensile stresses may exist.

Possible examination outcomes

- No relevant conditions identified
- One or more areas are identified with minor (short) crack-like indications
- One or more areas are identified with large (long) crack-like indications
- One or more areas are identified with missing material

Methodology and Data Requirements

The general analytical methodology to be used for acceptance criteria for the baffle plates involves the following steps and inputs:



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- Confirmation of required loading and combination requirements and determination of the high stressed regions in the baffle plates
- Perform a linear-elastic fracture mechanics (LEFM) evaluation to determine the critical crack size using the MRP-211 fracture toughness values
    - LEFM analysis will be used to determine the applied stress intensity factors (SIFs) for a through- thickness flawed configuration with normal/upset condition loads
    - Crack growth analysis is planned to be performed considering the normal/upset condition stresses and IASCC growth for a 10-year inspection interval
    - Critical crack sizes based on the degraded fracture toughness value and stresses under typical operating and LOCA conditions will be determined
    - Additionally, the minimum time to critical crack size will be determined
- If the above efforts are unsuccessful in concluding continued functionality, a bypass analysis may be performed to justify that sufficient DNB exists in the degraded condition
- A VT-1, ET, or UT examination may be needed to further characterize the flaws or to determine if flaws are emanating from the location where missing material may be identified
- An operability evaluation to operate at least one cycle based on possible inspection results for the plates should be performed
- An evaluation of the consequences of leaving cracked plates securely in place during an inspection or replacement campaign should be performed

The VT-3 exam acceptance criteria are “no relevant conditions.” Relevant conditions for the B&W Items of interest are defined in Section 5 of MRP-227-1A. These relevant conditions constitute discontinuities or imperfections (e.g., loose or missing parts, distortion, corrosion, wear, etc.), which are readily detectable with the VT-3 technique. Note that the VT-3 exam method meets the requirements of ASME B&PV Code Section XI, as specified and conditioned by 10 CFR 50.55a, and the MRP-228 Inspection Standard.

Input information needed includes:

- Identification of the most likely locations of cracking in the plates
- Identification of what visual examination indications are considered rejectable and would require additional examination and evaluation

Analytical efforts could be performed on a generic basis.

The licensee is required to submit the LEFM analysis for the baffle plates under normal/upset condition loads, considering IASCC, within one year of the inspection of the Primary component item for NRC to determine whether review and approval are needed if the inspection results indicate aging not meeting the acceptance criteria in Table 5-1 of MRP-227.

Existing Documentation

CLB loadings (normal and faulted condition) are available

- One appropriate model is available in EPRI Report *Models of Irradiation-Assisted Stress Corrosion Cracking of Austenitic Stainless Steels in Light Water Reactor Environments, Volume 2: Disposition Curves Application* [46], which provides the technical basis for ASME Code Case N-889 [52].



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## A.1.10 Locking Devices and Locking Welds of Baffle-to-Former and Internal Baffle-to-Baffle Bolts

### B11. Core Barrel Assembly

Locking devices, including locking welds, of baffle-to-former bolts and internal baffle-to-baffle bolts

There is a potential for irradiation-assisted stress corrosion cracking (IASCC) and irradiation embrittlement (IE) of the locking devices or welds for the baffle (baffle-to-former and internal baffle-to-baffle) bolting. There is also the potential for failure by overload for these locking devices and welds due to slip between the bolts and plates (see MRP-231). Past failure history exists with cracked and missing internal baffle-to-baffle bolt locking devices and cracked locking device welds in B&W-design units.

Component Item Function

The baffle-to-baffle bolts and the baffle-to-former bolts are subjected to flow-induced vibratory loads and significant thermal stresses due to the gamma heating in the core region. Both of these phenomena tend to cause bolt preload relaxation and a forcing function to unthread the bolts. The locking devices and welds serve to prevent unthreading and to capture the bolt head in the event that the bolt severs. They have marginal impact on maintaining bolt preload.

The locking devices and welds are not normally loaded unless the bolt is broken or loose. Loading of the locking devices and welds could also occur due to the slip between the bolt and plate. The locking devices and welds have no core support safety function.

Observable Effects

A visual (VT-3) examination of the bolt locking devices and welds is to be performed. Subsequent visual (VT-3) examinations are to be performed.

Cracking of the locking devices or welds is the main concern and they are to be examined to identify if any are separated, distorted, loose, broken, or missing.

Possible examination outcomes

- No relevant conditions identified
- One or a few bolt locking devices (up to 1% or 11) are identified with damage or are missing
- More than 1% (or 11) bolt locking devices are identified with damage or are missing

Methodology and Data Requirements

The VT-3 exam acceptance criteria are “no relevant conditions.” Relevant conditions for the B&W Items of interest are defined in Section 5 of MRP-227-1A. These relevant conditions constitute discontinuities or imperfections (e.g., loose or missing parts, distortion, corrosion, wear, etc.), which are readily detectable with the VT-3 technique. Note that the VT-3 exam method meets the requirements of ASME B&PV Code Section XI, as specified and conditioned by 10 CFR 50.55a, and the MRP-228 Inspection Standard.



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Existing Documentation

Inspection results from damaged and missing locking devices and welds from the CR-3 visual examinations can be used for development of an inspection standard, although additional efforts for other locking devices and welds will also be required.



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## A.1.11 Flow Distributor Bolts and Locking Devices

### B12. Flow Distributor Assembly

Flow distributor (FD) bolts and their locking devices

There is a potential for intergranular stress corrosion cracking (IGSCC) of Alloy A-286 and Alloy X-750 bolting. Past B&W failure history exists with the original Alloy A-286 bolt materials in B&W-design units and with applications of Alloy X-750 material within the nuclear industry (in general). Currently, there are no known UT indications of any of the replacement bolts (Alloy A-286 or Alloy X-750) in the operating B&W-design units. There are two instances of a UT indication or suspect indication where an original Alloy X-750 bolt is still in service in the operating B&W fleet (installed at TMI-1 only).

Component Item Function

The FD bolts fasten the flange on the flow distributor to the lower grid assembly. The joint also clamps the lower grid support plate in place between the bottom of the lower grid assembly and a ledge on the ID of the flow distributor. A clamp ring spans the gap between the bottom mating face of the lower grid assembly and the top of the support plate, providing the compressive force holding the support plate in place. The FD bolts do not have a core support function.

Observable Effects

A volumetric examination (UT) of the bolts and a visual (VT-3) examination of the bolt locking devices are to be performed. Subsequent visual (VT-3) examinations of the locking devices and ultrasonic (UT) examinations of the bolts are to be performed.

Cracking of the bolts is the main concern and the locking devices are to be examined to identify if any are damaged, distorted, loose, broken, or missing.

PWROG work evaluated the potential information that could be determined from only a visual examination of the bolt and locking devices.

Possible examination outcomes

*Bolts*

Any cracking is anticipated to be located at the head-to-shank area where the peak tensile stress exists (i.e., a SCF exists) and OE has shown high-strength bolting to crack. It is also possible that cracking may occur in the shank thread region where high tensile stress may exist.

- No relevant conditions identified
- Relevant conditions (i.e., crack-like indications, either completely cracked or partially cracked; or non-interpretable UT indications, such as no back wall reflection or multiple reflections with no crack-like indication that is most likely caused by a large or duplex grain size) are identified
    - One or a few bolts (exact number is unit-specific) are identified with relevant indications
    - More than a few bolts (exact number is unit-specific) are identified with relevant indications

*Locking Devices*

- No relevant conditions identified



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- One or two are identified with damage, distortion, or are missing
- More than two bolt locking devices are identified with damage, distortion, or are missing

Methodology and Data Requirements

*Bolts*

The general analytical methodology to be used for acceptance criteria for the bolts (i.e., acceptable bolt pattern) and to justify an inspection interval involves the following steps and inputs if relevant conditions have been identified in the FD bolts:

Methodology for ABPA

- A finite element model (FEM) is to be developed for the local geometry with contact conditions, pretension elements, loads and boundary conditions
- A thermal analysis is to be performed
    - Determines bolt temperatures and temperature gradients for normal operating conditions
- A structural analysis is to be performed in which failed bolts are inactive
    - Stress concentration factors are calculated to determine the peak stresses at the bolt head-to-shank fillet region under normal operating conditions
    - Analysis is performed for all loads and load combinations required for an ASME Code evaluation (stress limits for threaded structural fasteners in subsection NG and Appendix F)
    - An evaluation of joint stability (or, openness) is also to be performed
- Representative rejected FD bolts may be removed for laboratory testing to confirm the UT inspection results and IGSCC mechanism, or to identify any other failure mechanism(s)

NOTE: One alternative to removing representative UT rejected bolts for laboratory examination is to re-inspect all bolts at the next refueling outage if continued operation for one additional fuel cycle can be supported by a technical evaluation. Other possible options such as replacement of a predefined bolt pattern may also be pursued.

Methodology for using ABPA to justify a re-inspection interval

- Based on the results of FD bolt UT inspection and any laboratory test results, perform an evaluation to predict the number of failed bolts at the end of the re-inspection interval. The evaluation will use a bolt degradation model based on OE and/or mechanistic models of the degradation mechanism(s). An empirical and/or empirical and mechanistic model of the bolting degradation effect is developed from industry OE, plant-specific design information and/or test data. The model is calibrated to plant-specific bolt inspection results; or otherwise applies a bolt degradation model for similar plants after accounting for relevant differences, as appropriate. The changes to the peak stress at the bolt head-to-shank fillet region as a result of the identified failures should be included for evaluation of increased susceptibility to SCC.
- Incorporate the effect of future FD bolt failure into the operability evaluation and re-inspection requirement. The predicted number of failed bolts at the end of the re-inspection interval determined in the previous step will be used to project a statistically significant number of future bolting patterns (e.g. 10,000 samples) that may exist at the end of the re-inspection interval.



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- The projected statistically significant number of future bolting patterns (e.g. 10,000 samples) that may exist at the end of the re-inspection interval will be compared to the results of the ABPA to justify at least a 95% probability of an acceptable bolt pattern at the end of the re-inspection interval. It is anticipated that this method will be used to demonstrate a 10-year re-inspection interval. Re-inspection intervals beyond 10-years may be justifiable in certain circumstances.

The licensee shall submit the plant-specific analysis for this component item within one year after the inspection if the results do not meet the acceptance criteria in MRP-227.

*Locking Devices*

The VT-3 exam acceptance criteria are "no relevant conditions." Relevant conditions for the B&W Items of interest are defined in Section 5 of MRP-227-1A. These relevant conditions constitute discontinuities or imperfections (e.g., loose or missing parts, distortion, corrosion, wear, etc.), which are readily detectable with the VT-3 technique. Note that the VT-3 exam method meets the requirements of ASME B&PV Code Section XI, as specified and conditioned by 10 CFR 50.55a, and the MRP-228 Inspection Standard.

Data Requirements

Analytical efforts for the FD bolts could be performed on a generic basis (for all units except TMI-1), although unit-specific analyses could decrease the conservatism for some units.

Existing Documentation

A generic flow distributor bolt stress analysis was developed.



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## A.1.12 Guide Block Dowel Welds

### B13. Lower Grid Assembly

Alloy X-750 dowel-to-guide block welds (all units except DB-1)

There is a potential for primary water stress corrosion cracking (PWSCC) of the dowel-to-guide block welds. No known OE exists for cracking in these welds, but OE with cracking of Alloy 82/182 weld materials in PWRs does exist.

Component Item Function

These welds serve as loose part prevention devices and are not structural. Small cracks in the locking weld are acceptable since the locking function can be maintained as long as any part of the weld is present.

The dowel-to-guide block welds have no core support function.

Observable Effects

A visual (VT-3) examination of the dowel-to-guide block welds is to be performed. Subsequent visual (VT-3) examinations are to be performed.

Loss of the locking function of the weld is the main concern and therefore the dowel-to-guide block welds are to be examined to identify if any are separated or missing or if a dowel is missing.

Possible examination outcomes

- No relevant conditions exist
- A single weld is observed to be damaged or partially missing
- Several welds are observed to be damaged or partially missing
- A single weld or dowel is missing
- Several welds or dowels are missing
- A guide block is misaligned or is missing
- Several guide blocks are misaligned or are missing

Methodology and Data Requirements

The VT-3 exam acceptance criteria are “no relevant conditions.” Relevant conditions for the B&W Items of interest are defined in Section 5 of MRP-227-1A. These relevant conditions constitute discontinuities or imperfections (e.g., loose or missing parts, distortion, corrosion, wear, etc.), which are readily detectable with the VT-3 technique. Note that the VT-3 exam method meets the requirements of ASME B&PV Code Section XI, as specified and conditioned by 10 CFR 50.55a, and the MRP-228 Inspection Standard.

Existing Documentation

Unit-specific engineering evaluations and assessments (aka functionality analyses) have been performed



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## A.1.13 Shock Pad Bolts and Locking Devices

### B14. Lower Grid Assembly

Shock pad bolts and their locking devices (TMI-1 only)

There is a potential for intergranular stress corrosion cracking (IGSCC) of Alloy X-750 bolting. Past B&W failure history exists with applications of Alloy X-750 material within the nuclear industry (in general). Currently, there are two instances of a UT indication or suspect indication where an original Alloy X-750 bolt in service in an operating B&W-design unit (installed at TMI-1 only).

Component Item Function

The function of the lower grid shock pad bolts is to fasten the shock pads to the lower grid assembly, at TMI-1 they also function as lower thermal shield (LTS) bolts. The LTS bolt joint acts as a restraint to vertical and rotational motion of the bottom of the thermal shield, but does not act as a direct core support component and therefore does not have a core support safety function. Evaluation of the LTS bolt joint is based on 96 LTS bolts with no credit taken for strength of the SP bolts. Therefore, the only function to be maintained by the SP bolts is to keep the shock pads in place.

Observable Effects

A volumetric examination (UT) of the bolts and a visual (VT-3) examination of the bolt locking devices are to be performed. Subsequent visual (VT-3) examinations of the locking devices and ultrasonic (UT) examinations of the bolts are to be performed.

Cracking of the bolts is the main concern and the locking devices are to be examined to identify if any are damaged, distorted, loose, broken, or missing.

PWROG work evaluated the potential information that could be determined from only a visual examination of the bolt and locking devices.

Possible examination outcomes

*Bolts*

Any cracking is anticipated to be located at the head-to-shank area where the peak tensile stress exists (i.e., a SCF exists) and OE has shown high-strength bolting to crack. It is also possible that cracking may occur in the shank thread region where high tensile stress may exist.

- No relevant conditions identified
- Relevant conditions (i.e., crack-like indications, either completely cracked or partially cracked; or non-interpretable UT indications, such as no back wall reflection or multiple reflections with no crack-like indication that is most likely caused by a large or duplex grain size) are identified
    - If one bolt is missing on a shock pad, it is regarded as relevant since there is no redundancy to hold the pad in place
    - If two bolts on any individual shock pad are identified with relevant conditions, the shock pad could become a loose part and may not be in place in the event of a core drop accident



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*Locking Devices*

- No relevant conditions identified
- One or two are identified with damage, distortion, or are missing
- More than two bolt locking devices are identified with damage, distortion, or are missing

Methodology and Data Requirements

*Bolts*

The general analytical methodology to be used for acceptance criteria for the bolts involves the following steps and inputs if relevant conditions have been identified in the lower grid shock pad bolts:

- If two bolts on any individual shock pad are identified with relevant indications, it is an indication that the shock pad may become loose and will not be in place to carry a core drop. A structural evaluation is to be performed to determine if remaining pads can carry the core drop load or if the load can be carried without the shock pad in place
- If one of two bolts has an indication, an analysis is to be performed to assess loads on the remaining bolt and its potential for future failure
    - Loads on this bolt will include those evaluated as part of modeling of the lower thermal shield joint as described for LTS bolts
    - Stress concentration factors are calculated to determine the peak stresses at the bolt head-to-shank fillet region under normal operating conditions
    - Structural evaluation will be performed to determine peak stress in remaining shock pad bolts for use in assessing potential for future bolt failure
- Representative rejected lower grid shock pad bolts may be removed for laboratory testing to confirm the UT results and IGSCC mechanism, or to identify any other failure mechanism(s)

NOTE: One alternative to removing representative UT rejected bolts for laboratory examination is to re-inspect all bolts at the next refueling outage if continued operation for one additional fuel cycle can be supported by a technical evaluation. Other possible options such as bolt replacement may also be pursued.

- Based on the results of lower grid shock pad bolt UT inspection and any laboratory test results, perform an evaluation to assess future lower grid shock pad bolt failure potential. The changes to the peak stress at the bolt head-to-shank fillet region as a result of the identified failures should be included for evaluation of increased susceptibility to SCC
- Incorporate the effect of future lower grid shock pad bolt failure into the operability evaluation and re-inspection requirement

The licensee shall submit the plant-specific analysis for this component item within one year after the inspection if the results do not meet the acceptance criteria in MRP-227.

*Locking Devices*

The VT-3 exam acceptance criteria are “no relevant conditions.” Relevant conditions for the B&W Items of interest are defined in Section 5 of MRP-227-1A. These relevant conditions constitute discontinuities or imperfections (e.g., loose or missing parts, distortion, corrosion, wear, etc.), which are readily detectable with the VT-3 technique. Note that the VT-3 exam method meets the requirements of ASME B&PV Code Section XI, as specified and conditioned by 10 CFR 50.55a, and the MRP-228 Inspection Standard.



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Existing Documentation

Minimal information is readily available.



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## A.1.14 Incore Monitoring Instrumentation Guide Tube Spiders and Welds

### B15. Incore Monitoring Instrumentation (IMI) Guide Tube Assembly

IMI guide tube spiders

IMI guide tube spider-to-lower grid rib section welds

The IMI guide tube spiders and welds are subject to both thermal aging and irradiation embrittlement, which if a flaw would be present and they are subjected to loading that exceeds the materials degraded fracture toughness, such a condition could potentially lead to cracking. Recent examinations at B&W units have identified potential crack-like indications in a small number of guide tube spider legs.

Component Item Function

The purpose of the spiders is to provide lateral restraint for the IMI guide tubes and the purpose of the spider fillet welds is to hold the spiders in place. The spiders and spider-to-lower grid rib section welds have a minimal role in actually supporting the IMI guide tubes and do not provide a core support function. Although it is unlikely, significant degradation of the IMI guide tube spiders could result in misalignment of the guide tubes with the IMI nozzles and subsequent insertion or retraction of the in-core monitoring instrumentation.

Observable Effects

A visual (VT-3) examination of the IMI guide tube spiders is to be performed. Subsequent visual (VT-3) examinations are to be performed.

The IMI guide tube spiders are being examined to detect fractured or missing spider arms or any separated or missing welds.

Possible examination outcomes:

- No relevant conditions identified
- One or more spider arms are identified to be fractured
- One or more spider arms are identified with a separated weld
- One or more spider arms are identified to be missing
- One or more spider arms are identified with a missing weld
- One or more spider arms are not aligned with the lower grid fuel assembly screw (as viewed from the top)

Methodology and Data Requirements

The general analytical methodology to be used for acceptance criteria for the IMI guide tube spiders involves the following steps and inputs:

- Prepare an evaluation, such as a flow-induced vibration (FIV) analysis, to insure that the IMI spider will have adequate structural integrity to prevent instability and loss of function
- A VT-1, ET, or UT examination may be needed to determine if flaws are emanating from the location where missing material may be identified



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The VT-3 exam acceptance criteria are “no relevant conditions.” Relevant conditions for the B&W Items of interest are defined in Section 5 of MRP-227-1A. These relevant conditions constitute discontinuities or imperfections (e.g., loose or missing parts, distortion, corrosion, wear, etc.), which are readily detectable with the VT-3 technique. Note that the VT-3 exam method meets the requirements of ASME B&PV Code Section XI, as specified and conditioned by 10 CFR 50.55a, and the MRP-228 Inspection Standard.

The licensee shall submit the plant-specific analysis for this component item within one year after the inspection if the results do not meet the acceptance criteria in MRP-227.

Existing Documentation

- CLB loadings (normal and faulted condition) are available
- Unit-specific stress analyses have been performed



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## A.2 EXPANSION COMPONENT ITEMS

Acceptance criteria methodology and data requirements for each of the Expansion component items are summarized in this section. A separate sub-section is provided for each component item using the following format:

- Component item identification and background information.
- Component item function(s), including whether or not it has a core support safety function.
- Observable effect(s).
- Possible examination outcome(s).
- Methodology for development of acceptance criteria.
- Data requirements for development of acceptance criteria.
- Existing documents (e.g., PWROG or Framatome).



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## A.2.1 Upper Grid Fuel Assembly Support Pad Dowel Welds

### B13.1. Upper Grid Assembly

Alloy X-750 dowel-to-upper grid fuel assembly support pad welds (except DB-1, which will be addressed on a unit-specific basis)

There is a potential for primary water stress corrosion cracking (PWSCC) of the dowel-to-upper grid fuel assembly support pad welds. No known OE of cracking exists for these welds in B&W-design units, but OE with cracking of Alloy 82/182 weld materials in PWRs does exist.

Component Item Function

These welds serve as loose part prevention devices and are not structural. Small cracks in the locking weld are acceptable since the locking function can be maintained as long as any part of the weld is present.

The upper grid fuel assembly support pads serve as guidance for loading and alignment of the fuel assemblies into the core. The upper grid fuel assembly support pads transfer the fuel spring load to the upper grid assembly. The dowels maintain the horizontal position of the fuel assembly support pads. There are minor horizontal operating loads on the pads due to cross flow in the fuel and fuel interference due to bowing. The upper grid fuel assembly support pads are assumed to be in the seismic load path from the fuel assembly to the upper grid. Therefore, the dowels provide a core support function.

Observable Effects

A visual (VT-3) examination of the dowel-to-upper grid fuel assembly support pad welds is to be performed. Subsequent visual (VT-3) examinations are to be performed.

Loss of the locking function of the weld is the main concern and therefore the dowel-to-upper grid fuel assembly support pad welds are to be examined to identify if any welds are separated or missing. The examination will also identify if a dowel is missing, or the support pad is misaligned (clearly out of perpendicularity).

Possible examination outcomes

- No relevant conditions exist
- A single weld is observed to be damaged or partially missing
- Several welds are observed to be damaged or partially missing
- A single weld and dowel is missing
- Several welds and dowels are missing
- A support pad is misaligned (clearly out of perpendicularity) or missing
- Several support pads are misaligned (clearly out of perpendicularity) or missing

Methodology and Data Requirements

The VT-3 exam acceptance criteria are “no relevant conditions.” Relevant conditions for the B&W Items of interest are defined in Section 5 of MRP-227-1A. These relevant conditions constitute discontinuities or imperfections (e.g., loose or missing parts, distortion, corrosion, wear, etc.), which are readily detectable with



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the VT-3 technique. Note that the VT-3 exam method meets the requirements of ASME B&PV Code Section XI, as specified and conditioned by 10 CFR 50.55a, and the MRP-228 Inspection Standard.

Existing Documentation

Minimal information is currently available.



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## A.2.2 Vent Valve Assembly Vent Valve Bodies

### B2.1. Vent Valve Assembly

Vent valve bodies

Cast austenitic stainless steel (CASS) vent valve bodies are potentially susceptible to thermal aging embrittlement, which if a flaw would be present and they are subjected to loading that exceeds the material's reduced fracture toughness, such a condition could potentially lead to cracking. There is no known confirmed cracking of CASS material in PWR reactor vessel internals applications.

Component Item Function

Vent valve assemblies are installed in the mounting rings in the core support shield. For all normal operating conditions, the vent valve is closed. In the event of a cold leg large break LOCA, the valve will open to vent steam generated in the core directly to the break, thus permitting the core to be flooded and adequately cooled after emergency core coolant has been supplied to the reactor vessel. Each vent valve assembly consists of a hinged disc, a valve body with sealing surfaces, two retaining rings and fasteners, and an alignment device (to maintain the correct orientation). Each vent valve assembly can be remotely handled as a unit for removal or installation. Vent valve component parts, including the disc, are designed to minimize the possibility of loss of parts to the reactor coolant system, and all operating fasteners include a positive locking device. The hinged disc includes a device for remote testing and verification of proper disc function. The external side of the disc is contoured to absorb the impact load of the disc on the RV inside wall without transmitting excessive impact loads to the hinge parts as a result of a LOCA.

Observable Effects

A visual (VT-3) examination of accessible surfaces of 100% of the vent valve bodies on the plenum side (core side) of the vent valve is to be performed. Subsequent visual (VT-3) examinations are to be performed.

The vent valve bodies are being examined to detect cracking; this includes the detection of fractured vent valve bodies or surface irregularities (e.g., damaged, grossly cracked, or missing portions) associated with the vent valve bodies.

Possible examination outcomes

- No relevant and/or reportable conditions identified
- Observation that the vent valve is not symmetric in the mounting ring
- Observation of a vent valve body incorrectly positioned
- Observation of a vent valve body with surface irregularities (e.g., damaged, grossly cracked, or missing portions)
- Observation of a fractured vent valve body

Methodology and Data Requirements

The general analytical methodology to be used for acceptance criteria for the vent valve bodies involves the following steps and inputs:

- Determine impact of observation(s) on future function of vent valve body and overall vent valve function and/or design configuration



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- Determine if the vent valve assembly can be left as-is (with observations) or if any actions (up to vent valve replacement) need to be taken for the vent valve (both vent valve body and overall vent valve) to perform its intended function

Existing Documentation

- Calculations of ferrite content have been performed for various vent valve bodies
- A statistical assessment of PWR RV internals CASS materials has been made for the PWROG (PWROG-15032-P, Revision 0-A).
- Several previous analyses of observations have been performed with varying results (vent valve left as-is versus vent valve replacement)
- Bypass leakage calculations are available
- Exercise load calculations are available
- Original design calculations, such as vent valve opening force calculations, are available

Note: In the event of expansion, as an alternative to inspecting the vent valve bodies, analysis using data from fabrication records and/or a statistical evaluation may be performed to justify that the vent valve bodies are not susceptible to thermal aging embrittlement.



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## A.2.3 Upper Thermal Shield Bolts and Locking Devices

### B7.1. Core Barrel Assembly

Upper thermal shield (UTS) bolts and their locking devices

There is a potential for intergranular stress corrosion cracking (IGSCC) of Alloy A-286 and Alloy X-750 bolting. Past B&W failure history exists with the original Alloy A-286 bolt materials in B&W-design units and with applications of Alloy X-750 material within the nuclear industry (in general). Currently, there are no known UT indications of with any of the replacement bolts (Alloy A-286 or Alloy X-750) in the operating B&W-design units. There are two instances of a UT indication or suspect indication where an original Alloy X-750 bolt is still in service in the operating B&W fleet (installed at TMI-1 only).

Component Item Function

The UTS bolts fasten a split restraint and shim block to the core barrel. The UTS bolts do not have a core support function.

Observable Effects

A volumetric examination (UT) of the bolts and a visual (VT-3) examination of the bolt locking devices are to be performed. Subsequent visual (VT-3) examinations of the locking devices and ultrasonic (UT) examinations of the bolts are to be performed.

Cracking of the bolts is the main concern and the locking devices are to be examined to identify if any are damaged, distorted, loose, broken, or missing.

PWROG work evaluated the potential information that could be determined from only a visual examination of the bolt and locking devices.

Possible examination outcomes

*Bolts*

Any cracking is anticipated to be located at the head-to-shank area where the peak tensile stress exists (i.e., a SCF exists) and OE has shown high-strength bolting to crack. It is also possible that cracking may occur in the shank thread region where high tensile stress may exist.

- No relevant conditions identified
- Relevant conditions (i.e., crack-like indications, either completely cracked or partially cracked; or non-interpretable UT indications, such as no back wall reflection or multiple reflections with no crack-like indication that is most likely caused by a large or duplex grain size) are identified
    - One or a few bolts (exact number is unit-specific) are identified with relevant indications
    - More than a few bolts (exact number is unit-specific) are identified with relevant indications

*Locking Devices*

- No relevant conditions identified
- One or two are identified with damage, distortion, or are missing
- More than two bolt locking devices are identified with damage, distortion, or are missing



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Methodology and Data Requirements

*Bolts*

The general analytical methodology to be used for acceptance criteria for the bolts involves the following steps and inputs if relevant conditions have been identified in the UTS bolts:

- If one or two of the three bolts on a restraint has an indication, an analysis is to be performed to assess loads on the remaining bolt(s) and its potential for future failure
    - Loads on this bolt will include those evaluated as part of modeling of the UTS joint as described for UTS bolts
    - Stress concentration factors are calculated to determine the peak stresses at the bolt head-to-shank fillet region under normal operating conditions
    - Structural evaluation will be performed to determine peak stress in remaining UTS bolts for use in assessing potential for future bolt failure
- If the structural evaluation concludes that one or more restraints with degraded UTS bolts may not perform their function, a structural evaluation is to be performed to determine if the remaining restraints are sufficient to perform the cumulative function of the restraints
- Representative rejected UTS bolts may be removed for laboratory testing to confirm the UT inspection results and IGSCC mechanism, or to identify any other failure mechanism(s)

NOTE: One alternative to removing representative UT rejected bolts for laboratory examination is to re-inspect all bolts at the next refueling outage if continued operation for one additional fuel cycle can be supported by a technical evaluation. Other possible options such as bolt replacement may also be pursued.

- Based on the results of UTS bolt UT inspection and any laboratory test results, perform an evaluation to assess future UTS bolt failure potential. The changes to the peak stress at the bolt head-to-shank fillet region as a result of the identified failures should be included for evaluation of increased susceptibility to SCC
- Incorporate the effect of future UTS bolt failure into the operability evaluation and re-inspection requirement

The licensee shall submit the unit-specific analysis for this component item within one year after the inspection of the linked Primary item when the results trigger the expansion criteria in MRP-227.

*Locking Devices*

The VT-3 exam acceptance criteria are “no relevant conditions.” Relevant conditions for the B&W Items of interest are defined in Section 5 of MRP-227-1A. These relevant conditions constitute discontinuities or imperfections (e.g., loose or missing parts, distortion, corrosion, wear, etc.), which are readily detectable with the VT-3 technique. Note that the VT-3 exam method meets the requirements of ASME B&PV Code Section XI, as specified and conditioned by 10 CFR 50.55a, and the MRP-228 Inspection Standard.

Existing Documentation

A stress analysis has been performed generically and an acceptable failure pattern analysis has been performed on a unit-specific basis.



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## A.2.4 Surveillance Specimen Holder Tube Studs/Nuts or Bolts and Locking Devices (DB-1 only)

### B7.2. Core Barrel Assembly

Surveillance specimen holder tube (SSHT) bolts and their locking devices (DB-1 only)

There is a potential for intergranular stress corrosion cracking (IGSCC) of Alloy X-750 bolting. Past failure history exists with the Alloy X-750 material within the nuclear industry. Currently, there are no known failures with any of the replacement bolts in the operating B&W-design units. There are two instances of a UT indication or suspect indication where an original Alloy X-750 bolt is still in service in the operating B&W fleet (installed at TMI-1 only).

Component Item Function

The SSHT bolts fasten the surveillance specimen holder tubes to the thermal shield. Failure would result in loosening or dropping of the holder tube to the bottom of the vessel. The SSHT bolts do not have a core support safety function.

Observable Effects

A volumetric examination (UT) of the bolts and a visual (VT-3) examination of the bolt locking devices are to be performed. Subsequent visual (VT-3) examinations of the locking devices and ultrasonic (UT) examinations of the bolts are to be performed.

Cracking of the bolts is the main concern and the locking devices are to be examined to identify if any are damaged, distorted, loose, broken, or missing.

PWROG work evaluated the potential information that could be determined from only a visual examination of the bolt and locking devices.

Possible examination outcomes

*Bolts*

Any cracking is anticipated to be located at the head-to-shank area where the peak tensile stress exists (i.e., a SCF exists) and OE has shown high-strength bolting to crack. It is also possible that cracking may occur in the shank thread region where high tensile stress may exist.

- No relevant conditions identified
- Relevant conditions (i.e., crack-like indications, either completely cracked or partially cracked; or non-interpretable UT indications, such as no back wall reflection or multiple reflections with no crack-like indication that is most likely caused by a large or duplex grain size) are identified
    - No more than one bolt per bracket identified with a relevant indication
    - More than one bolt per bracket identified with a relevant indication

*Locking Devices*

- No relevant conditions identified
- One or two are identified with damage, distortion, or are missing
- More than two bolt locking devices are identified with damage, distortion, or are missing



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Methodology and Data Requirements

*Bolts*

The general analytical methodology to be used for acceptance criteria for the bolts involves the following steps and inputs if relevant conditions have been identified in the SSHT bolts:

- A finite element model (FEM) is to be developed for the local geometry with contact conditions, pretension elements, loads and boundary conditions
- A thermal analysis is to be performed
    - Determines bolt temperatures and temperature gradients for normal operating conditions
- A structural analysis is to be performed in which failed bolts are inactive
    - Stress concentration factors are calculated to determine the peak stresses at the bolt head-to-shank fillet region or stud/nut thread region under normal operating conditions
    - Analysis is performed for all loads and load combinations required for an ASME Code evaluation (stress limits for threaded structural fasteners in subsection NG and Appendix F)
- Representative rejected SSHT bolts may be removed for laboratory testing to confirm the UT inspection results and IGSCC mechanism, or to identify any other failure mechanism(s)

NOTE: One alternative to removing representative UT rejected bolts for laboratory examination is to re-inspect all bolts at the next refueling outage if continued operation for one additional fuel cycle can be supported by a technical evaluation. Other possible options such as bolt replacement may also be pursued.

- Based on the results of SSHT bolt UT inspection and any laboratory test results, perform an evaluation to assess future SSHT bolt failure potential. The changes to the peak stress at the bolt head-to-shank fillet region as a result of the identified failures should be included for evaluation of increased susceptibility to SCC
- Incorporate the effect of future SSHT bolt failure into the operability evaluation and re-inspection requirement

The licensee shall submit the plant-specific analysis for this component item within one year after the inspection of the linked Primary item when the results trigger the expansion criteria in MRP-227.

*Locking Devices*

The VT-3 exam acceptance criteria are “no relevant conditions.” Relevant conditions for the B&W Items of interest are defined in Section 5 of MRP-227-1A. These relevant conditions constitute discontinuities or imperfections (e.g., loose or missing parts, distortion, corrosion, wear, etc.), which are readily detectable with the VT-3 technique. Note that the VT-3 exam method meets the requirements of ASME B&PV Code Section XI, as specified and conditioned by 10 CFR 50.55a, and the MRP-228 Inspection Standard.

Existing Documentation

Minimal information is available.



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## A.2.5 Core Barrel Cylinder

### B10.1. Core Barrel Assembly

Core barrel cylinder (including vertical and center circumferential seam welds)

The core barrel cylinder is subject to irradiation embrittlement, which, if a flaw were present and subjected to loading that exceeds the materials degraded fracture toughness, could potentially lead to cracking. There is no known history of OE for cracking of core barrel cylinder in B&W reactor vessel internals applications; indications have been observed at a PWR (CE) unit.

Component Item Function

Degradation of the core barrel cylinder could result in increased core bypass flow and a reduction in margin to DNB, but would not impede rod insertion. The core barrel supports the fuel assemblies, lower grid, flow distributor, and incore instrument guide tubes. The core barrel and its flanged connections at the ends that bolt to the core support shield assembly and lower grid assembly, respectively, directly perform the core support function and provide a flow envelope for the core that limits core bypass flow, particularly during a loss-of-coolant-accident (LOCA). If the core barrel cylinder or any other part of the core support assembly were to fail, causing a postulated drop of the core, the RV core guide lugs would limit the drop to assure that the control elements, even in the fully withdrawn position, remain engaged in the fuel assemblies.

Observable Effects

The core barrel cylinder is inaccessible and is addressed by analysis. No examination is required.

Methodology and Data Requirements

Since this component item is addressed by analysis and inspection is not required, methodology to address inspection findings is not applicable.

The licensee shall submit the plant-specific analysis, replacement schedule, or justification for some other alternative process for this component item within one year after the inspection of the linked Primary item when the results trigger the expansion criteria in MRP-227.



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## A.2.6 Former Plates

**B10.2. Core Barrel Assembly**

Former plates

Former plates are subject to irradiation embrittlement, which if a flaw would be present and they are subjected to loading that exceeds the materials degraded fracture toughness; such a condition could potentially lead to cracking. There is no known history of OE for cracking of former plate material in PWR reactor vessel internals applications.

Component Item Function

The former plates do not have a direct core support safety function; however, they do have a safety function to help maintain the structural integrity of the core barrel assembly during operating conditions.

Observable Effects

The former plates are mostly inaccessible without disassembly. No examination is required.

The top portion of the uppermost former plate is partially accessible from above the core barrel assembly and portions of the remaining former plates are partially accessible through openings in the core barrel assembly that could potentially be visually (VT-3) examined if techniques were required to be developed, but substantial development efforts would be needed.

Methodology and Data Requirements

Since this component item is addressed by analysis and inspection is not required, methodology to address inspection findings is not applicable.

The licensee shall submit the plant-specific analysis, replacement schedule, or justification for some other alternative process for this component item within one year after the inspection of the linked Primary item when the results trigger the expansion criteria in MRP-227.



*** This record was final approved on 08/31/2023 13:45:20. (This statement was added by the PRIME system upon its validation)

## A.2.7 Baffle-to-Baffle Bolts

### B9.1. Core Barrel Assembly

Baffle-to-baffle bolts

There is a potential for failure in the form of cracking of baffle-to-baffle bolting (both internal and external) to occur mainly from irradiation-assisted stress corrosion cracking (IASCC), but also as a result of irradiation embrittlement, irradiation creep/stress relaxation (leading to fatigue and wear), or overload (from a prying effect, for internal baffle-to-baffle bolts). Past failure history exists with baffle-to-former bolt materials (Type 316CW and Type 347) and core barrel-to-former bolt materials (Alloy X-750 and Type 316Ti CW) in non-B&W-design units. Currently, of the Type 304 bolts in the operating B&W-design units, there is one known UT indication (which was a baffle-to-former bolt). CR-3 has reportedly observed possibly failed internal baffle-to-baffle bolts, but confirmation has never been made.

Component Item Function

The core barrel assembly consists of the core barrel cylinder, former plates, and baffle plates connected by bolted joints that include: (1) core barrel-to-former bolts (CF bolts), (2) baffle-to-former bolts (BF bolts), and (3) baffle-to-baffle bolts (BB bolts). The core barrel assembly supports the fuel assemblies, lower grid, flow distributor, and incore instrument guide tubes. The baffle plates, former plates, and their joints (including the bolts) do not have a core support function and are categorized as internals structures. The primary function of the baffle plates, former plates, and their bolted connections is to provide a flow envelope surrounding the core. Also, since they are bolted to the core barrel cylinder, the baffle plates and former plates will produce a small increase on the stiffness and natural frequencies of the core barrel assembly.

The CF and BF bolts have the function of maintaining structural integrity of the baffle and former portion of the structural assembly and thus of maintaining flow geometry during normal operation. For faulted events, a small number of the CF and BF bolts are needed to restrain the baffle so that a coolable core geometry is maintained. The BB bolts are not required for these functions but rather serve to minimize gaps between baffle plates. The BB bolts therefore do not have a core support function.

Observable Effects

The baffle-to-baffle bolts are addressed by analysis. No examinations are required

Methodology and Data Requirements

Since this component item is addressed by analysis and inspection is not required, methodology to address inspection findings is not applicable.

The licensee shall submit the plant-specific analysis, replacement schedule, or justification for some other alternative process for this component item within one year after the inspection of the linked Primary item when the results trigger the expansion criteria in MRP-227.



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## A.2.8 Core Barrel-to-Former Bolts

**B9.2. Core Barrel Assembly**

Core barrel-to-former bolts

There is a potential for failure in the form of cracking of core barrel-to-former bolting to occur mainly from irradiation-assisted stress corrosion cracking (IASCC), but also as a result of irradiation embrittlement, irradiation creep/stress relaxation (leading to fatigue and wear). Past failure history exists with baffle-to-former bolt materials (Type 316CW and Type 347) and core barrel-to-former bolt materials (Alloy X-750 and Type 316Ti CW) in non-B&W-design units. Currently, of the bolts (Type 304) in the operating B&W-design units, there is one known UT indication (which was a baffle-to-former bolt). CR-3 has reportedly observed possibly failed baffle-to-baffle bolts, but confirmation has never been made.

Component Item Function

The core barrel assembly consists of the core barrel cylinder, former plates, and baffle plates connected by bolted joints that include: (1) core barrel-to-former bolts (CF bolts), (2) baffle-to-former bolts (BF bolts), and (3) baffle-to-baffle bolts (BB bolts). The core barrel assembly supports the fuel assemblies, lower grid, flow distributor, and incore instrument guide tubes. The baffle plates, former plates, and their joints (including the bolts) do not have a core support function and are categorized as internals structures. The primary function of the baffle plates, former plates, and their bolted connections is to provide a flow envelope surrounding the core. Also, since they are bolted to the core barrel cylinder, the baffle plates and former plates will produce a small increase on the stiffness and natural frequencies of the core barrel assembly.

The CF and BF bolts have the function of maintaining structural integrity of the baffle and former portion of the structural assembly and thus of maintaining flow geometry during normal operation. For faulted events, a small number of the CF and BF bolts are needed to restrain the baffle so that a coolable core geometry is maintained.

Observable Effects

The core barrel-to-former bolts are addressed by analysis. No examinations are required.

Methodology and Data Requirements

Since this component item is addressed by analysis and inspection is not required, methodology to address inspection findings is not applicable.

The licensee shall submit the plant-specific analysis, replacement schedule, or justification for some other alternative process for this component item within one year after the inspection of the linked Primary item when the results trigger the expansion criteria in MRP-227.



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## A.2.9 Locking Devices for External Baffle-to-Baffle and Core Barrel-to-Former Bolts

### B11.1. Core Barrel Assembly

Locking devices, including locking welds, of the external baffle-to-baffle bolts and core barrel-to-former bolts

There is a potential for irradiation embrittlement of the locking devices or welds for the external baffle-to-baffle and core barrel-to-former bolting, which if a flaw would be present and they are subjected to loading that exceeds the materials degraded fracture toughness, such a condition could potentially lead to cracking.

Component Item Function

The locking devices and welds are not normally loaded unless the bolt is broken or loose. Loading of the locking devices and welds could also occur due to the slip between the bolt and plate. The locking devices and welds have no core support function.

Observable Effects

These locking devices are addressed by analysis. No examinations are required.

Methodology and Data Requirements

Since this component item is addressed by analysis and inspection is not required, methodology to address inspection findings is not applicable.

The licensee shall submit the plant-specific analysis, replacement schedule, or justification for some other alternative process for this component item within one year after the inspection of the linked Primary item when the results trigger the expansion criteria in MRP-227.



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## A.2.10 Lower Grid Fuel Assembly Support Pad Items

### B15.1. 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

Lower grid fuel assembly support pad items are subject to irradiation embrittlement, which if a flaw would be present and they are subjected to loading that exceeds the materials degraded fracture toughness, such a condition could potentially lead to cracking. There is no known OE for cracking of these lower grid fuel assembly support pad items. However, a recent examination has identified a linear indication in the material of a lower grid rib section. The exact nature of that indication has not yet been determined.

Component Item Function

The cap screw locking welds serve as loose part prevention devices and are not structural. Small cracks in the locking weld are acceptable since the locking function can be maintained as long as any part of the weld is present. The lower grid fuel assembly support pads serve as guidance for loading of the fuel into the core and provide both vertical and lateral support for the fuel assembly. The lower grid fuel assembly support pads are attached to the lower grid by the cap screws and maintain their alignment and lateral support capability through the Alloy X-750 dowels.

Observable Effects

A visual (VT-3) examination of the lower grid fuel assembly support pad items is to be performed. Subsequent visual (VT-3) examinations are to be performed.

Loss of the lower grid fuel assembly support pad is the main concern and therefore the various items are to be examined to identify if any are separated or missing welds, if a dowel, cap screw, or support pad is missing, or if the support pad is misaligned (clearly out of perpendicularity).

Possible examination outcomes

- No relevant conditions exist
- A single weld is separated or missing
- Several welds are separated or missing
- A single dowel is missing
- Several dowels are missing
- A support pad is misaligned (clearly out of perpendicularity) or missing
- Several support pads are misaligned (clearly out of perpendicularity) or missing
- A single cap screw or locking weld is missing
- Several cap screws or locking welds are missing

Methodology and Data Requirements

The VT-3 exam acceptance criteria are “no relevant conditions.” Relevant conditions for the B&W Items of interest are defined in Section 5 of MRP-227-1A. These relevant conditions constitute discontinuities



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or imperfections (e.g., loose or missing parts, distortion, corrosion, wear, etc.), which are readily detectable with the VT-3 technique. Note that the VT-3 exam method meets the requirements of ASME B&PV Code Section XI, as specified and conditioned by 10 CFR 50.55a, and the MRP-228 Inspection Standard.

Existing Documentation

Minimal information is currently available.



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## A.2.11 Lower Grid Rib Section

### B10.3. Lower Grid Assembly

Lower grid rib section

The lower grid rib section is subject to irradiation embrittlement, which if a flaw would be present and it is subjected to loading that exceeds the materials degraded fracture toughness; such a condition could potentially lead to cracking. A small linear indication was observed in the lower grid rib section just below the spider-to-lower grid rib section weld toe at one B&W unit as part of the MRP-227 inspections of the IMI guide tube spider and spider-to-lower grid rib section welds.

Component Item Function

The lower grid assembly, which includes the lower grid rib section, provides alignment and support for the fuel assemblies, supports the core barrel assembly and flow distributor, and aligns the incore monitoring instrumentation (IMI) guide tubes with the fuel assembly instrument tubes. The lower grid rib section is a disk through which 177 squares are machined out, leaving a grid with "ribs." The square holes align with the fuel assembly locations in the core. There are additional holes about the periphery of the disk to permit a small bypass flow of reactor coolant up behind the baffle plates in the core barrel.

Observable Effects

A visual (VT-3) examination of 100% of the accessible surfaces of the lower grid rib section heat-affected zone (HAZ) adjacent to the spider-to-lower grid rib section welds is to be performed. Subsequent visual (VT-3) examinations are to be performed.

If relevant flaw(s) are identified, follow-on examination by VT-1, ET, or UT may need to be performed to confirm and characterize the length or both length and depth of the observation(s).

The lower grid rib section is being examined to detect cracking, including the detection of readily detectable cracking.

Possible Examination Outcomes

- No relevant and/or reportable conditions identified
- One or more lower grid rib section HAZ location(s) adjacent to the spider-to-lower grid rib section welds are identified with one or more minor (short) crack-like indicator(s)
- One or more lower grid rib section HAZ location(s) adjacent to the spider-to-lower grid rib section welds are identified with one or more large (long) crack-like indication(s)
- One or more lower grid rib section HAZ location(s) adjacent to the spider-to-lower grid rib section welds are identified with missing material

Methodology and Data Requirements

The general analytical methodology to be used for acceptance criteria for the lower grid rib section involves the following steps and inputs:

- Record length or length and depth of crack-like indication
- Perform appropriate fracture mechanics analysis
- Determine future inspection interval based on observed condition

Existing Documentation



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Minimal documentation currently exists for the lower grid rib section.

- A lower bound fracture toughness value for irradiated austenitic stainless steels has been documented (see MRP-175, Revision 1)
- Irradiated stainless steel CGRs are available; One appropriate model is available in EPRI Report *Models of Irradiation-Assisted Stress Corrosion Cracking of Austenitic Stainless Steels in Light Water Reactor Environments, Volume 2: Disposition Curves Application* [46], which provides the technical basis for ASME Code Case N-889 [52].
- Unit-specific justification for a re-examination interval has been performed



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## A.2.12 Dowel to Lower Grid Fuel Assembly Support Pad Dowel Welds

### B13.2. Lower Grid Assembly

Alloy X-750 dowel-to-lower grid fuel assembly support pad welds (except DB-1, which will be addressed on a unit-specific basis)

There is a potential for primary water stress corrosion cracking (PWSCC) of the dowel-to-lower grid fuel assembly support pad welds. No known OE of cracking exists for these welds in B&W-design units, but OE with cracking of Alloy 82/182 weld materials in PWRs does exist.

Component Item Function

These welds serve as loose part prevention devices and are not structural. Small cracks in the locking weld are acceptable since the locking function can be maintained as long as any part of the weld is present.

The lower grid fuel assembly support pads serve as guidance for loading of the fuel into the core and provide both vertical and lateral support for the fuel assembly. The lower grid fuel assembly support pads are attached to the lower grid by the cap screws and maintain their alignment and lateral support capability through the Alloy X-750 dowels. The dowel-to-lower grid fuel assembly support pad welds ensure the dowels remain engaged.

Observable Effects

A visual (VT-3) examination of the Alloy X-750 dowel-to-lower grid fuel assembly support pad welds is to be performed. Subsequent visual (VT-3) examinations are to be performed.

Loss of the locking function of the weld is the main concern and therefore the dowel-to-lower grid fuel assembly support pad welds are to be examined to identify if any welds are separated or missing. The examination will also identify if a dowel is missing, or the support pad is misaligned (clearly out of perpendicularity).

Possible examination outcomes

- No relevant conditions exist
- A single weld is observed to be damaged or partially missing
- Several welds are observed to be damaged or partially missing
- A single weld and dowel is missing
- Several welds and dowels are missing
- A support pad is misaligned (clearly out of perpendicularity) or missing
- Several support pads are misaligned (clearly out of perpendicularity) or missing

Methodology and Data Requirements

The VT-3 exam acceptance criteria are “no relevant conditions.” Relevant conditions for the B&W Items of interest are defined in Section 5 of MRP-227-1A. These relevant conditions constitute discontinuities or imperfections (e.g., loose or missing parts, distortion, corrosion, wear, etc.), which are readily detectable with the VT-3 technique. Note that the VT-3 exam method meets the requirements of ASME



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B&PV Code Section XI, as specified and conditioned by 10 CFR 50.55a, and the MRP-228 Inspection Standard.

Existing Documentation

Minimal information is currently available.



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## A.2.13 Lower Thermal Shield Bolts and Locking Devices

### B8.1. Lower Grid Assembly

Lower Thermal Shield (LTS) bolts (ANO-1, DB-1, and TMI-1) or studs/nuts (ONS-1, ONS-2, ONS-3) and their locking devices

There is a potential for intergranular stress corrosion cracking (IGSCC) of Alloy A-286 and Alloy X-750 bolting/studs/nuts. Past B&W failure history exists with the original Alloy A-286 bolt materials in B&W-design units and with applications of Alloy X-750 material within the nuclear industry (in general). Currently, there are no known UT indications with any of the replacement bolts (Alloy A-286 or Alloy X-750) in the operating B&W-design units. There are two instances of a UT indication or suspect indication where an original Alloy X-750 bolt is still in service in the operating B&W fleet (installed at TMI-1 only).

Component Item Function

The LTS studs/nuts or bolts fasten the thermal shield cylinder to the lower grid assembly. The LTS joint acts as a restraint to vertical and rotational motion of the bottom of the thermal shield, but does not act as a direct core support component and therefore does not have a core support safety function.

Observable Effects

A volumetric examination (UT) of the studs/nuts or bolts and a visual (VT-3) examination of the stud/nut or bolt locking devices are to be performed. Subsequent visual (VT-3) examinations of the locking devices and ultrasonic (UT) examinations of the studs/nuts or bolts are to be performed.

Cracking of the bolts or studs/nuts is the main concern and the locking devices are to be examined to identify if any are damaged, distorted, loose, broken, or missing.

The PWROG work also evaluated the potential information that could be determined from only a visual examination of the bolt and locking devices.

Possible examination outcomes

*Studs/Nuts or Bolts*

For the bolts, any cracking is anticipated to be located at the head-to-shank area where the peak tensile stress exists (i.e., a SCF exists) and OE has shown high-strength bolting to crack. For the studs/nuts or bolts, it is possible that cracking may occur in the shank or stud thread region where high tensile stress may exist.

- No relevant conditions identified
- Relevant conditions (i.e., crack-like indications, either completely cracked or partially cracked; or non-interpretable UT indications, such as no back wall reflection or multiple reflections with no crack-like indication that is most likely caused by a large or duplex grain size) are identified
    - One or a few studs/nuts or bolts (exact number is unit-specific) are identified with relevant indications
    - More than a few studs/nuts or bolts (exact number is unit-specific) are identified with relevant indications



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*Locking Devices*

- No relevant conditions identified
- One or two are identified with damage, distortion, or are missing
- More than two locking devices are identified with damage, distortion, or are missing

Methodology and Data Requirements

*Studs/Nuts or Bolts*

The general analytical methodology to be used for acceptance criteria for the studs/nuts or bolts involves the following steps and inputs if relevant conditions have been identified in the LTS studs/nuts or bolts:

- A finite element model (FEM) is to be developed for the local geometry with contact conditions, pretension elements, loads and boundary conditions
- A thermal analysis is to be performed
    - Determines stud/nut/bolt temperatures and temperature gradients for normal operating conditions
- A structural analysis is to be performed in which failed studs/nuts or bolts are inactive
    - Stress concentration factors are calculated to determine the peak stresses at the bolt head-to-shank fillet region or stud/nut thread region under normal operating conditions
    - Analysis is performed for all loads and load combinations required for an ASME evaluation (stress limits for threaded structural fasteners in subsection NG and Appendix F)
    - An evaluation of joint stability (or, openness) is also to be performed
- Representative rejected LTS studs/nuts or bolts may be removed for laboratory testing to confirm the UT inspection results and IGSCC mechanism, or to identify any other failure mechanism(s)

NOTE: One alternative to removing representative UT rejected studs/nuts or bolts for laboratory examination is to re-inspect all studs/nuts or bolts at the next refueling outage if continued operation for one additional fuel cycle can be supported by a technical evaluation. Other possible options such as replacement of a predefined stud/nut or bolt pattern may also be pursued.

- Based on the results of LTS stud/nut or bolt UT inspection and any laboratory test results, perform an evaluation to assess future LTS stud/nut or bolt failure potential. The changes to the peak stress at the bolt head-to-shank fillet region or stud/nut thread region as a result of the identified failures should be included for evaluation of increased susceptibility to SCC
- Incorporate the effect of future LTS stud/nut or bolt failure into the operability evaluation and re-inspection requirement

The licensee shall submit the plant-specific analysis for this component item within one year after the inspection of the linked primary item when the results trigger the expansion criteria in MRP-227.

*Locking Devices*

The VT-3 exam acceptance criteria are “no relevant conditions.” Relevant conditions for the B&W Items of interest are defined in Section 5 of MRP-227-1A. These relevant conditions constitute discontinuities or imperfections (e.g., loose or missing parts, distortion, corrosion, wear, etc.), which are readily detectable with the VT-3 technique. Note that the VT-3 exam method meets the requirements of ASME



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B&PV Code Section XI, as specified and conditioned by 10 CFR 50.55a, and the MRP-228 Inspection Standard.

Existing Documentation

A stress analysis has been performed generically and an acceptable failure pattern analysis has been performed on a unit-specific basis.



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# APPENDIX B
# FLOWCHARTS FOR B&W-DESIGN PRIMARY AND EXPANSION COMPONENT ITEM INSPECTIONS

## B.1 PRIMARY COMPONENT ITEMS

Flowcharts for each of the Primary component items are provided in this section. The flowcharts are a step-by-step guide for completing an evaluation of each component item. Refer to Appendix A for detailed text that forms the basis for the steps provided in each flowchart. A separate sub-section is provided for each component item flowchart.



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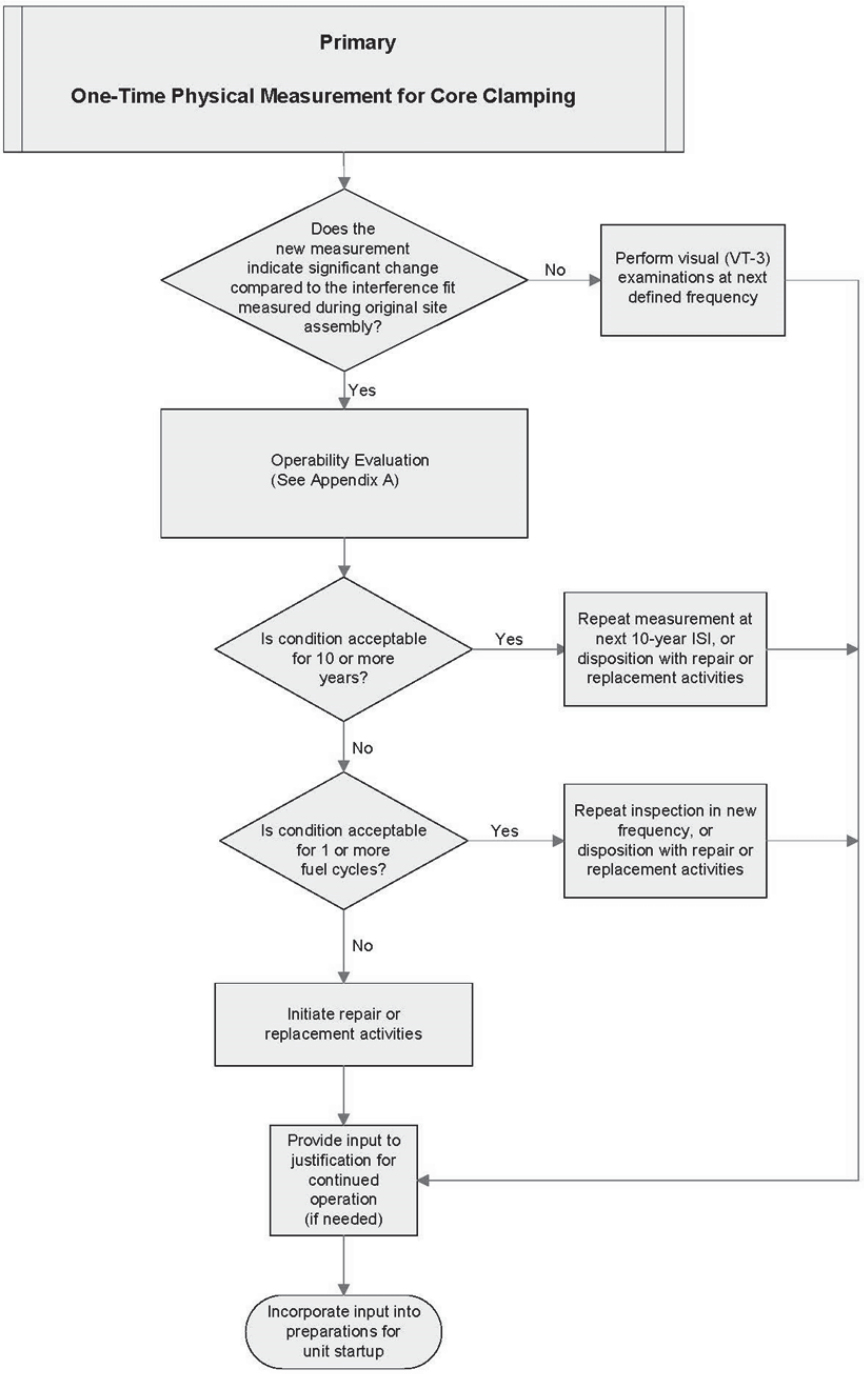
## B.1.1 Core Clamping Items (Physical Measurement)



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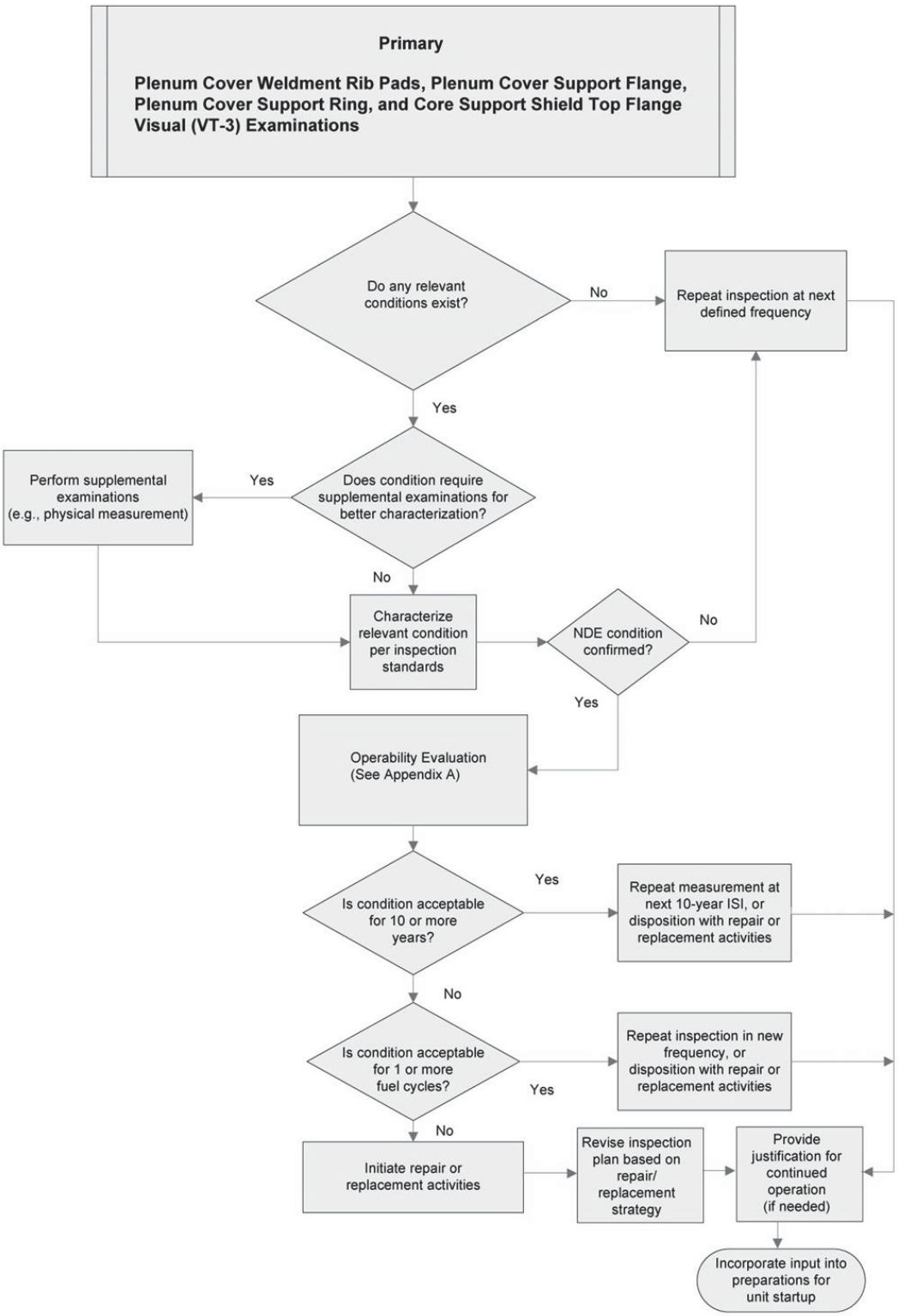
## B.1.2 Core Clamping Items (VT-3)



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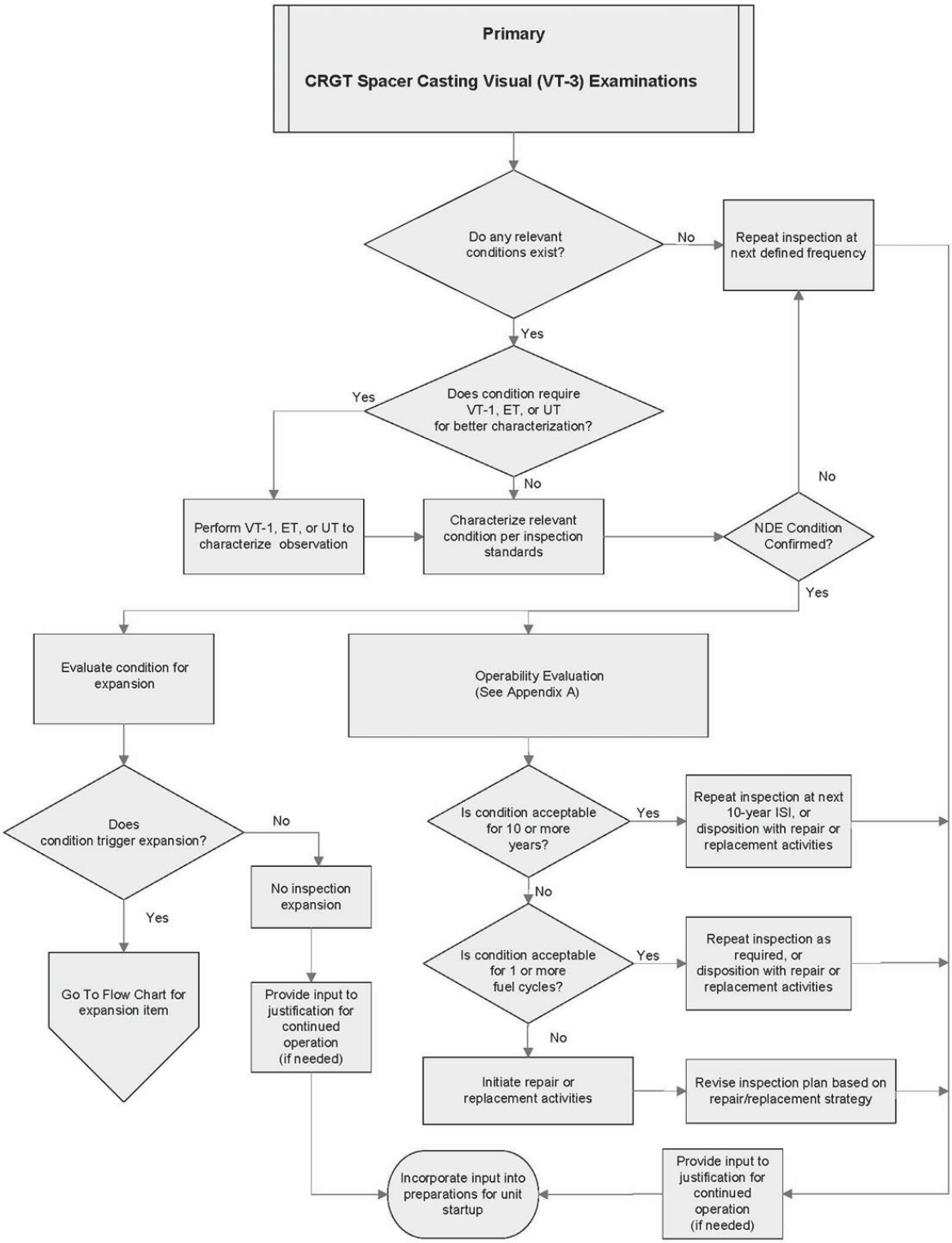
## B.1.3 Control Rod Guide Tube Spacer Castings



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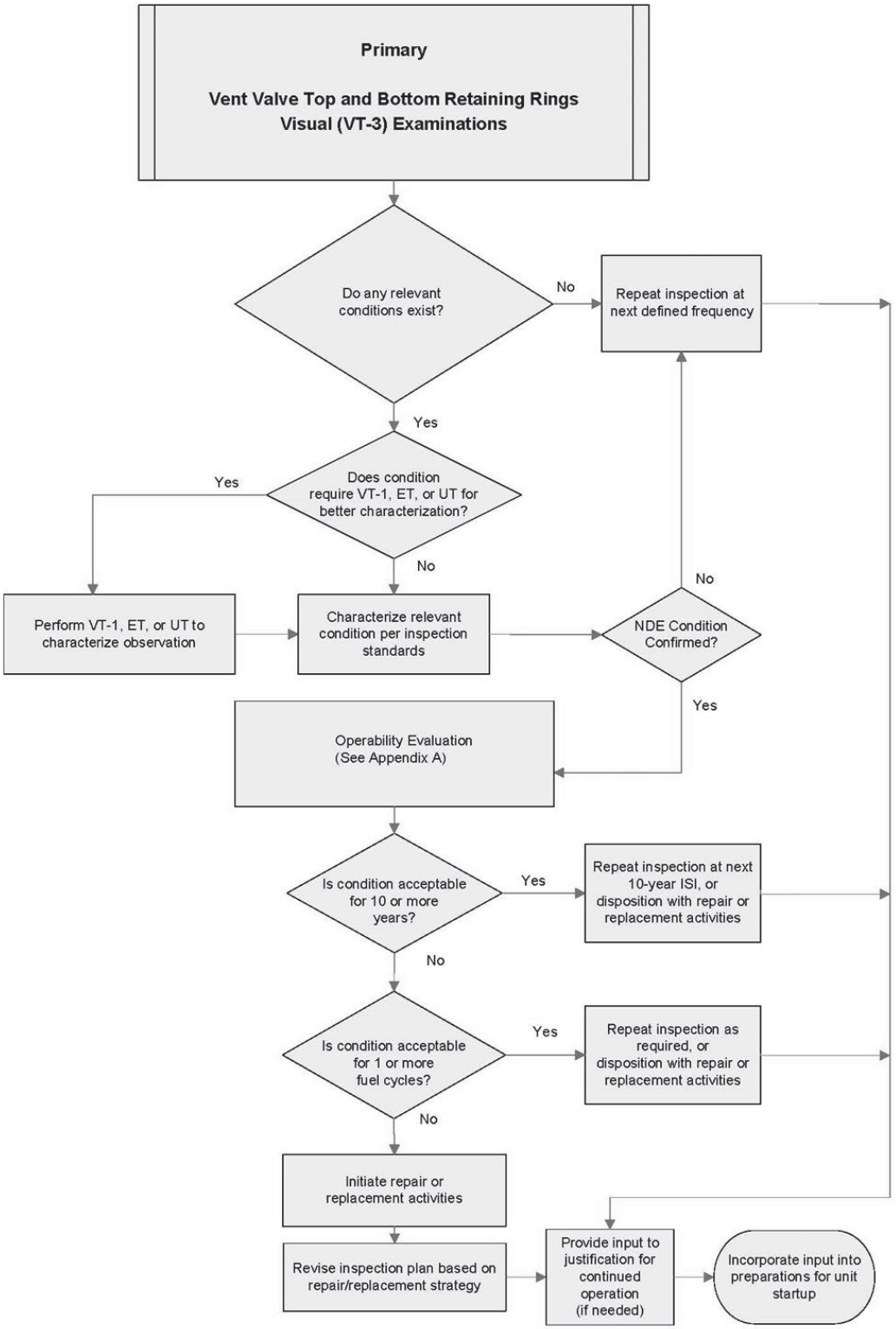
## B.1.4 Vent Valve Assembly Retaining Rings



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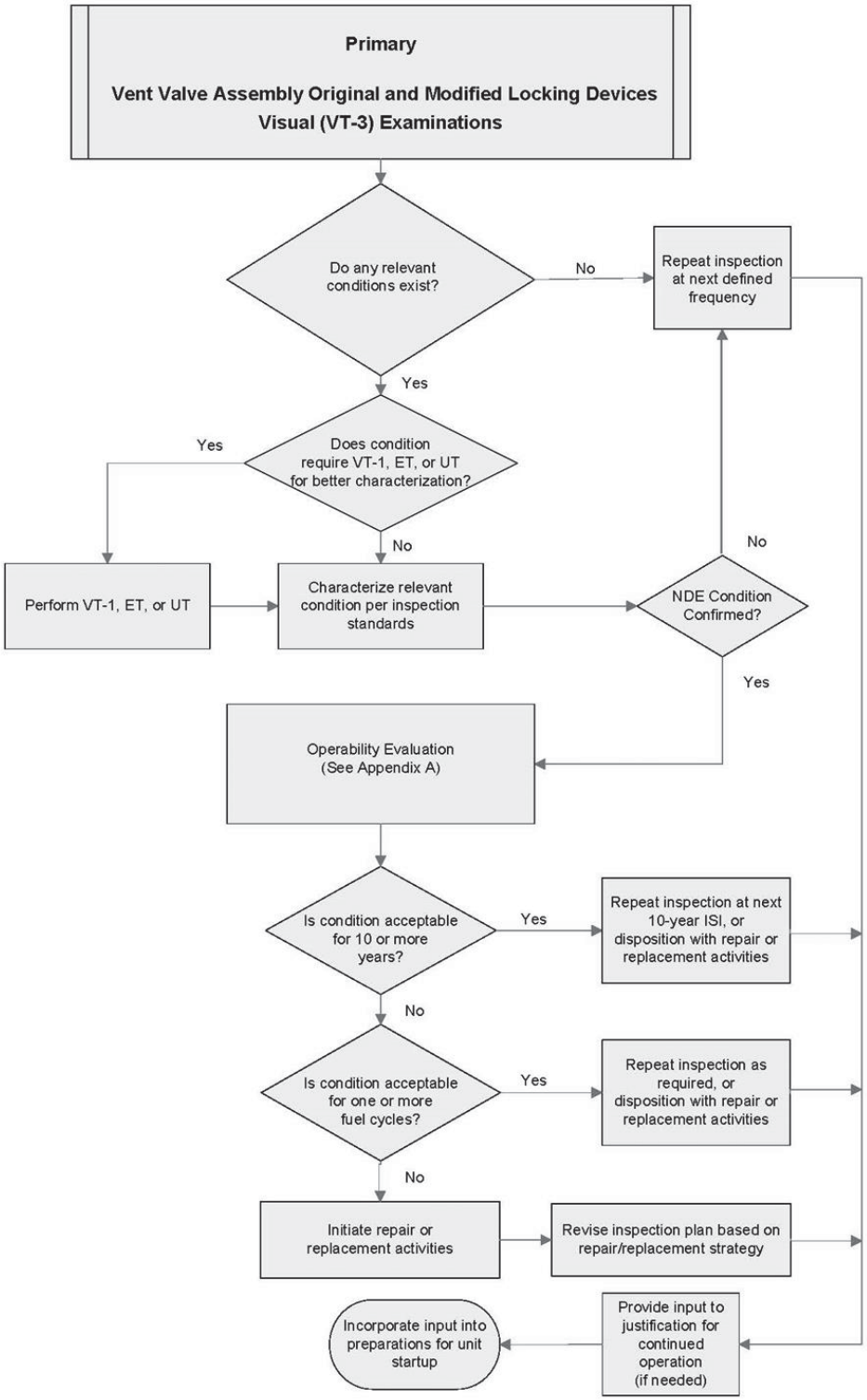
## B.1.5 Vent Valve Assembly Locking Devices



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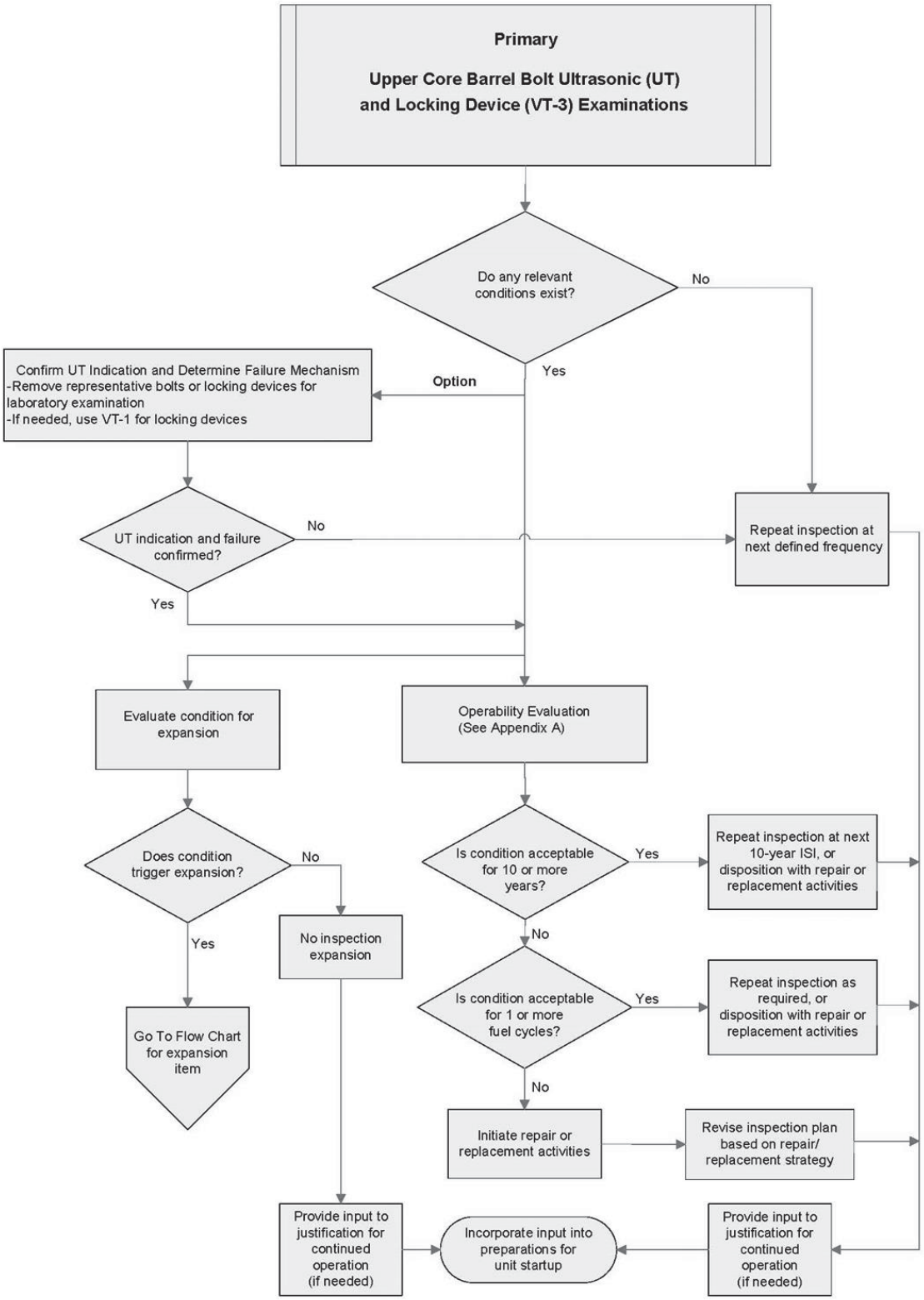
## B.1.6 Upper Core Barrel Bolts and Locking Devices



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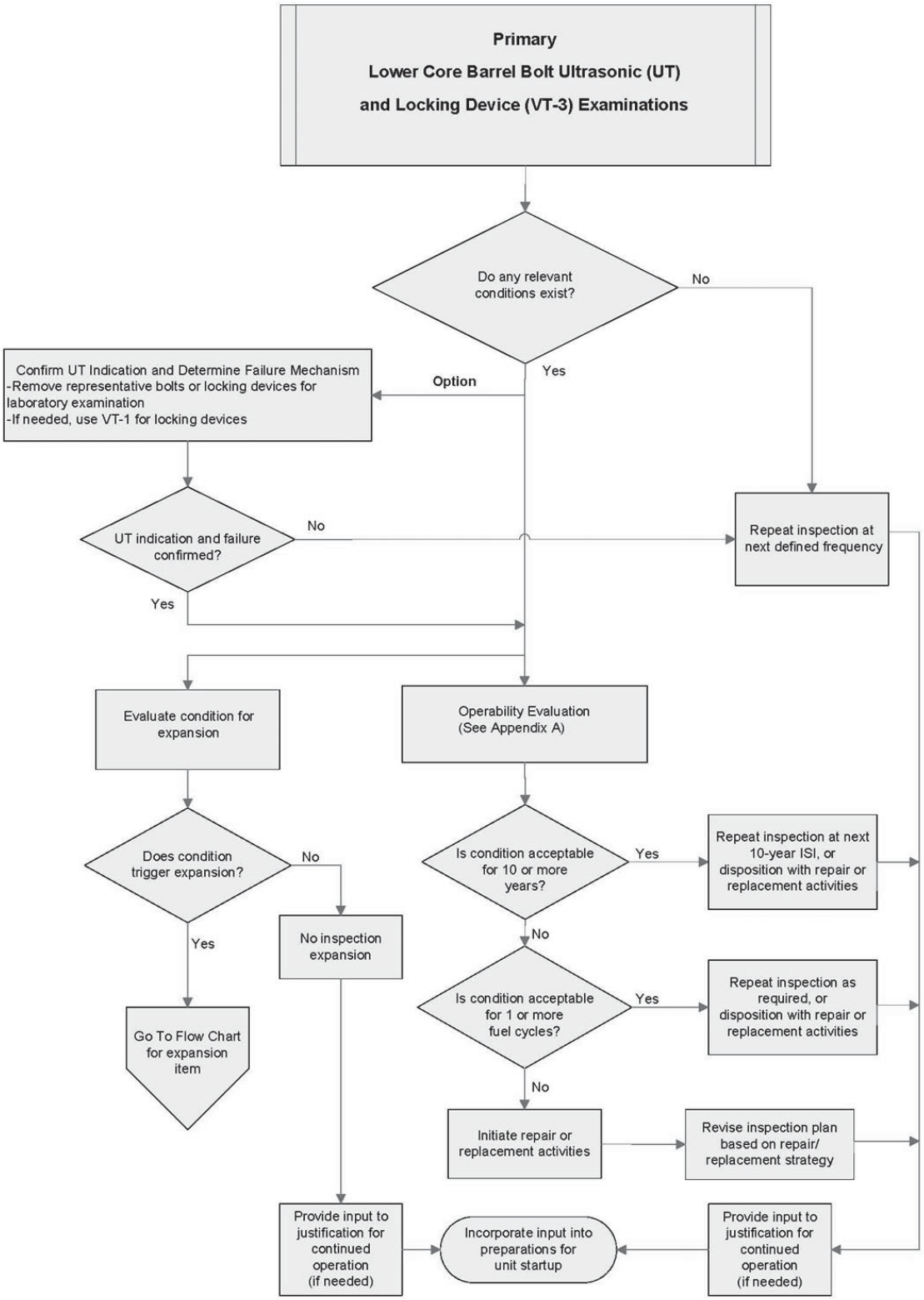
## B.1.7 Lower Core Barrel Bolts and Locking Devices



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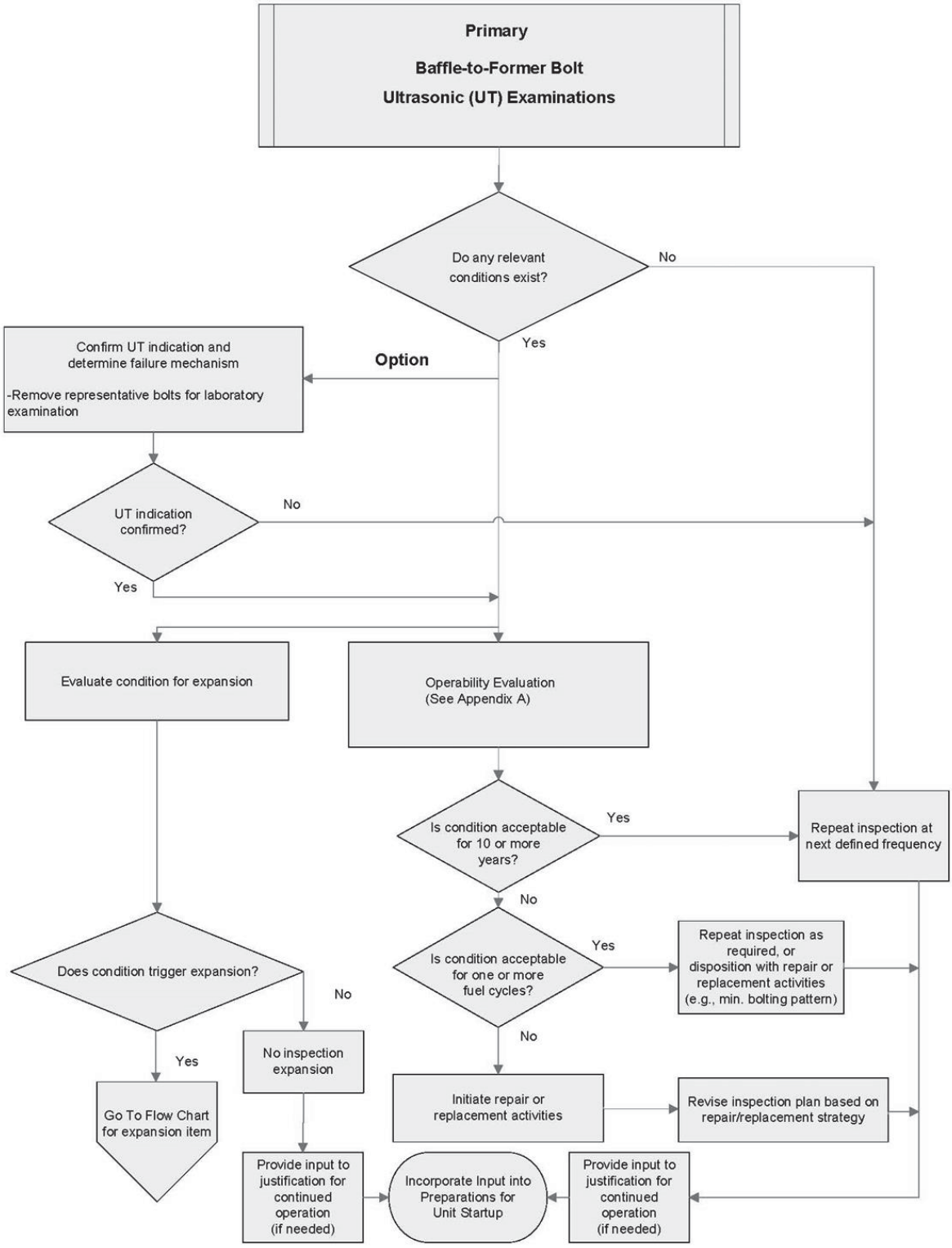
## B.1.8 Baffle-to-Former Bolts



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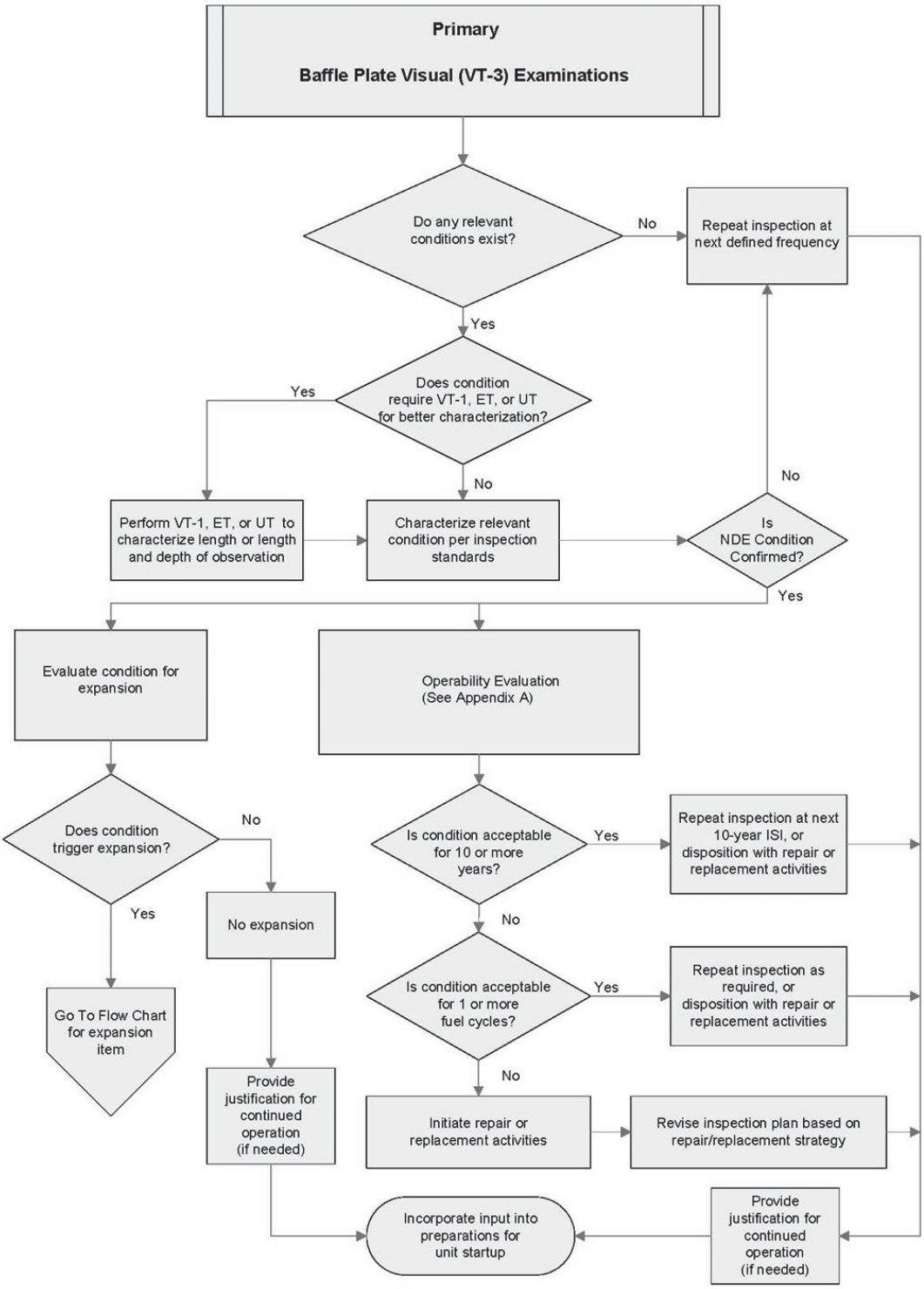
## B.1.9 Baffle Plates



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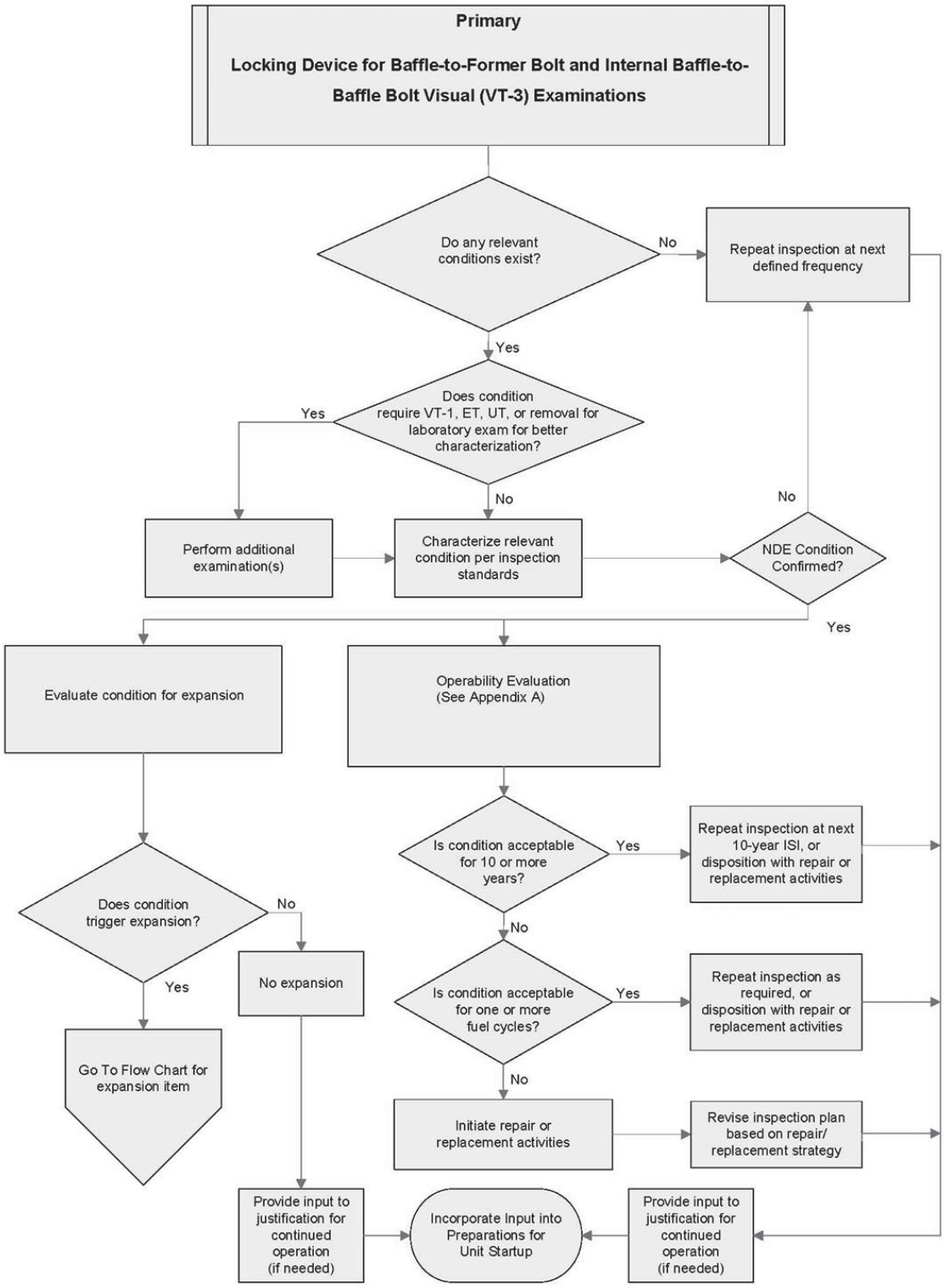
## B.1.10 Locking Devices of Baffle-to-Former and Internal Baffle-to-Baffle Bolts



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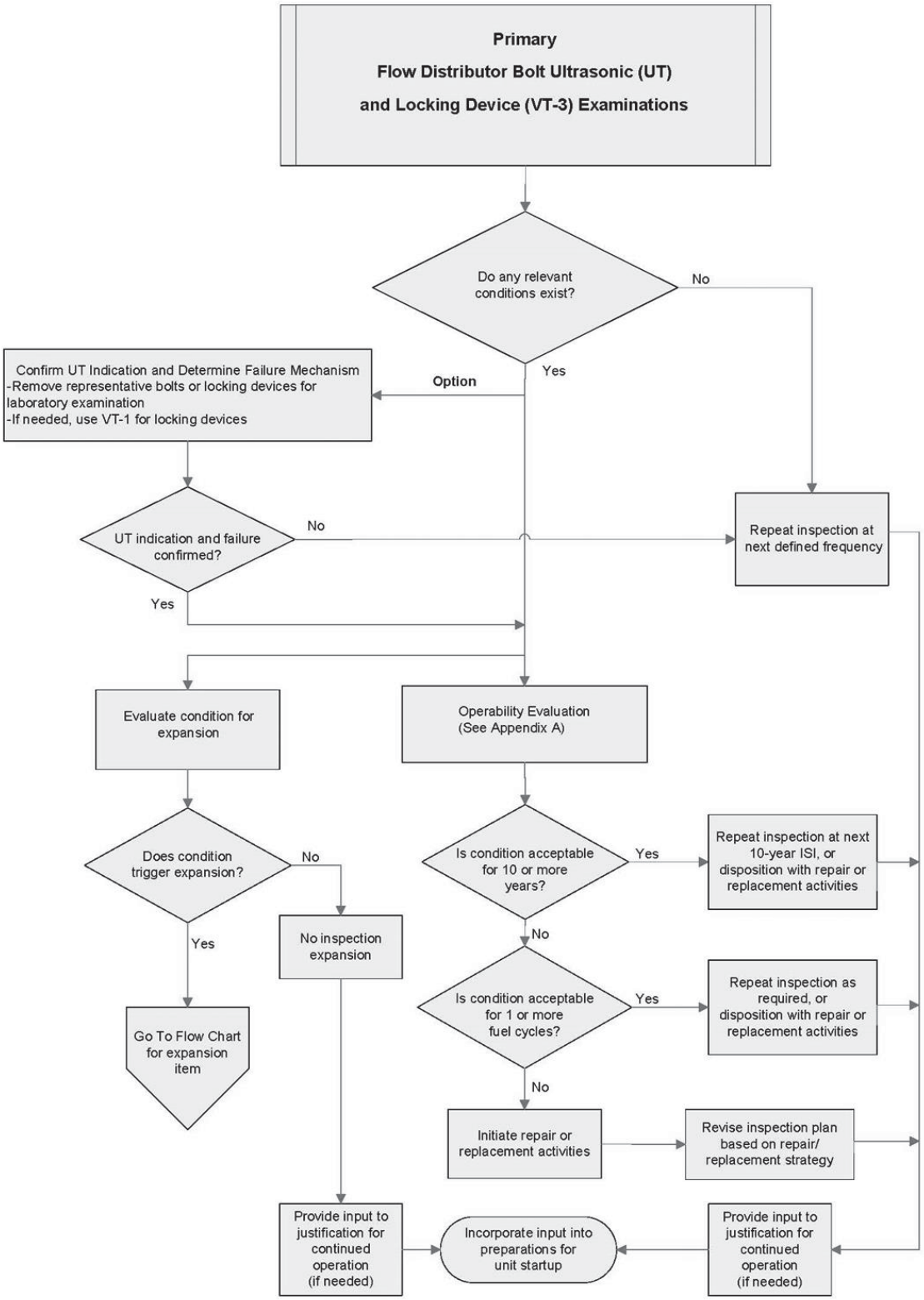
## B.1.11 Flow Distributor Bolts and Locking Devices



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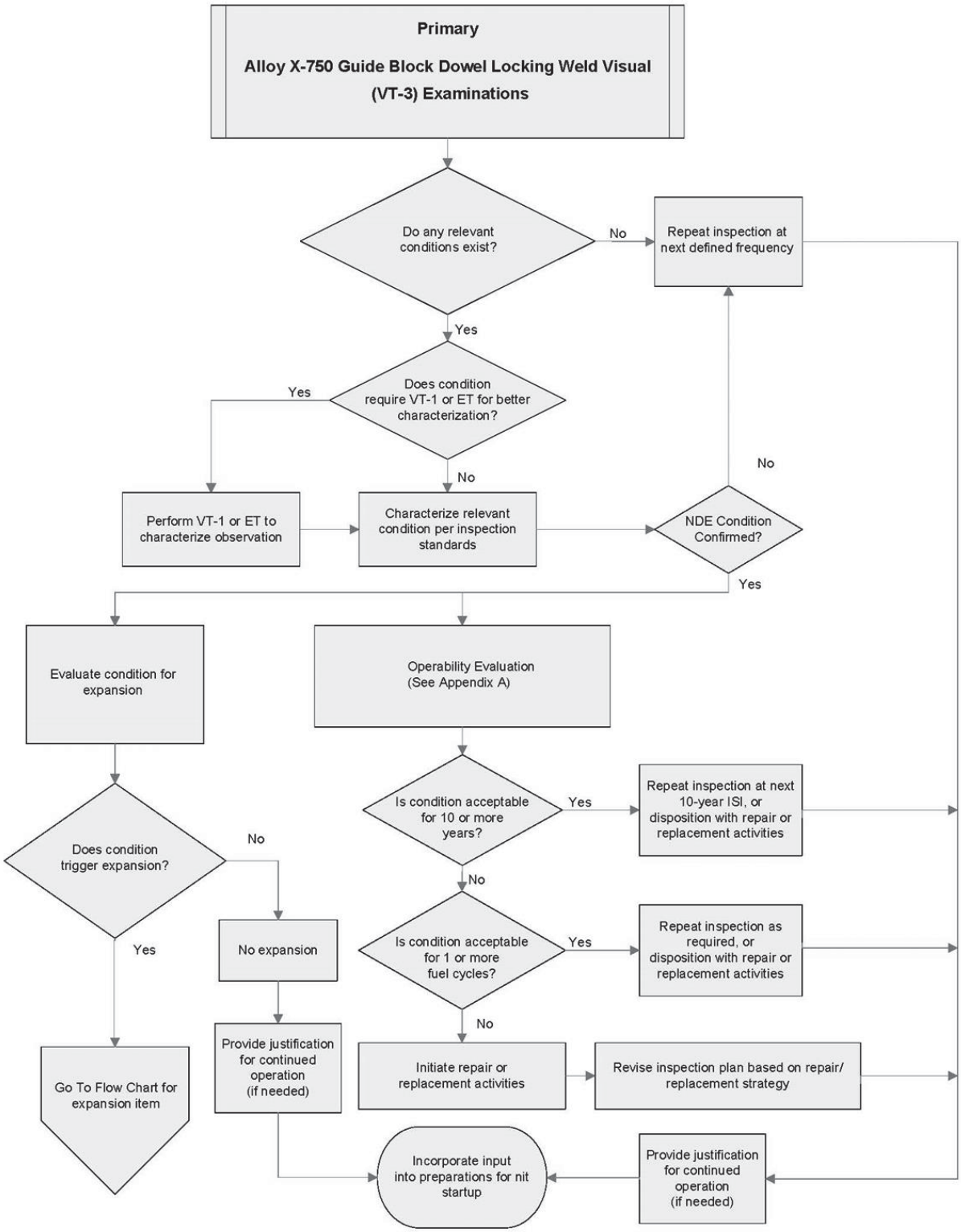
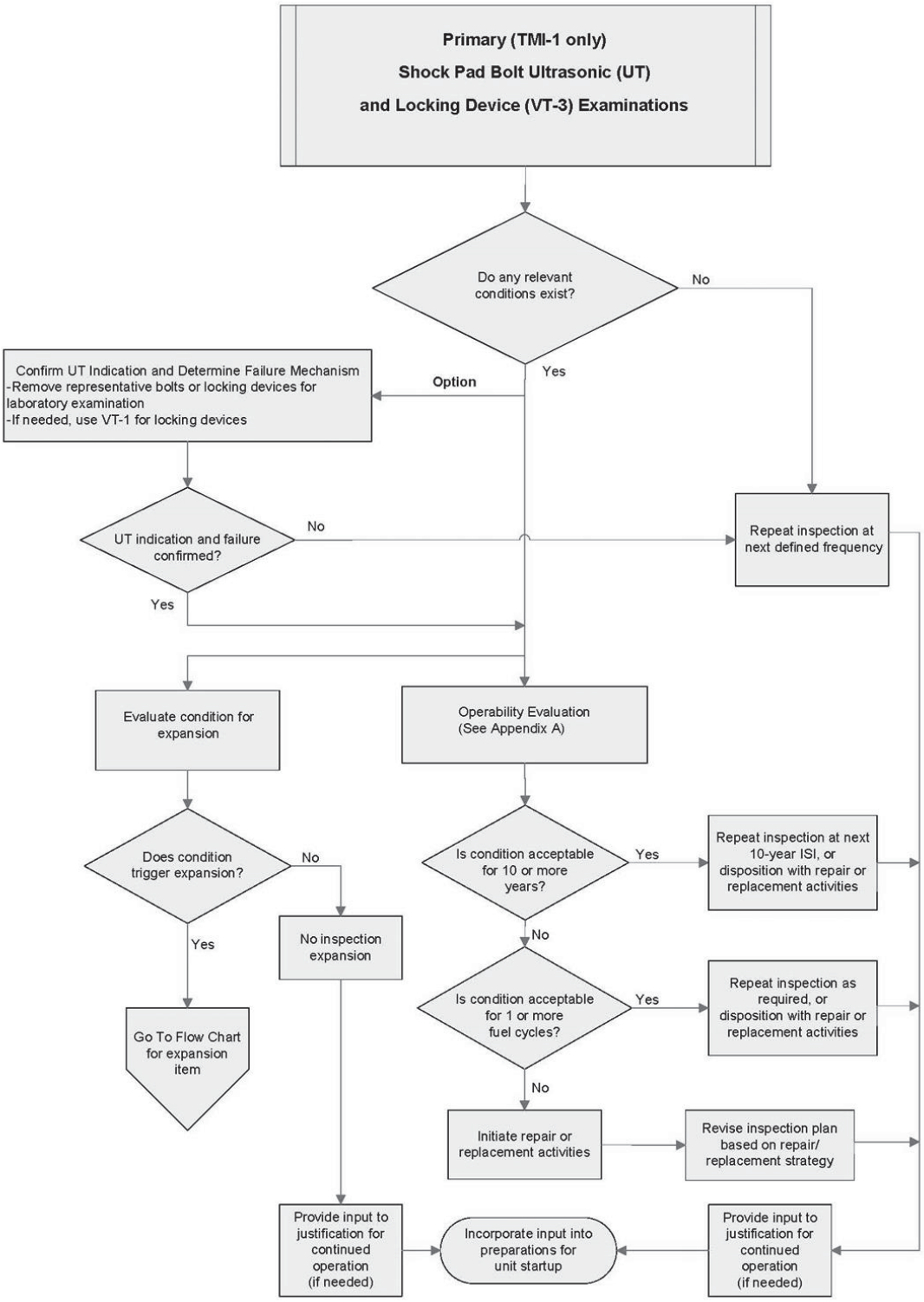
## B.1.12 Guide Block Dowel Locking Welds






*** This record was final approved on 08/31/2023 13:45:20. (This statement was added by the PRIME system upon its validation)

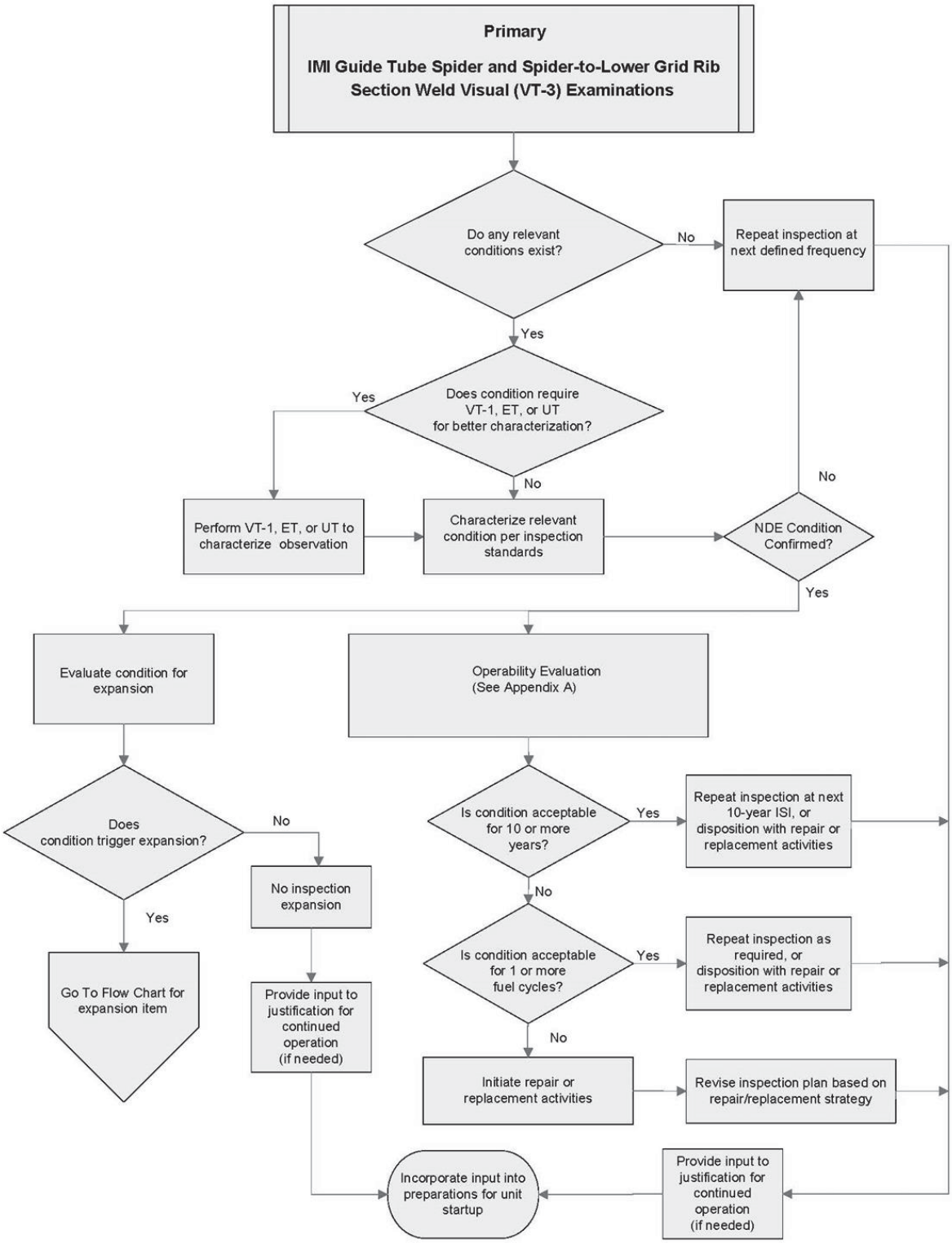
## B.1.13 Shock Pad Bolts and their Locking Devices (TMI-1)






*** This record was final approved on 08/31/2023 13:45:20. (This statement was added by the PRIME system upon its validation)

## B.1.14 Incore Monitoring Instrumentation Guide Tube Spiders and Welds






*** This record was final approved on 08/31/2023 13:45:20. (This statement was added by the PRIME system upon its validation)

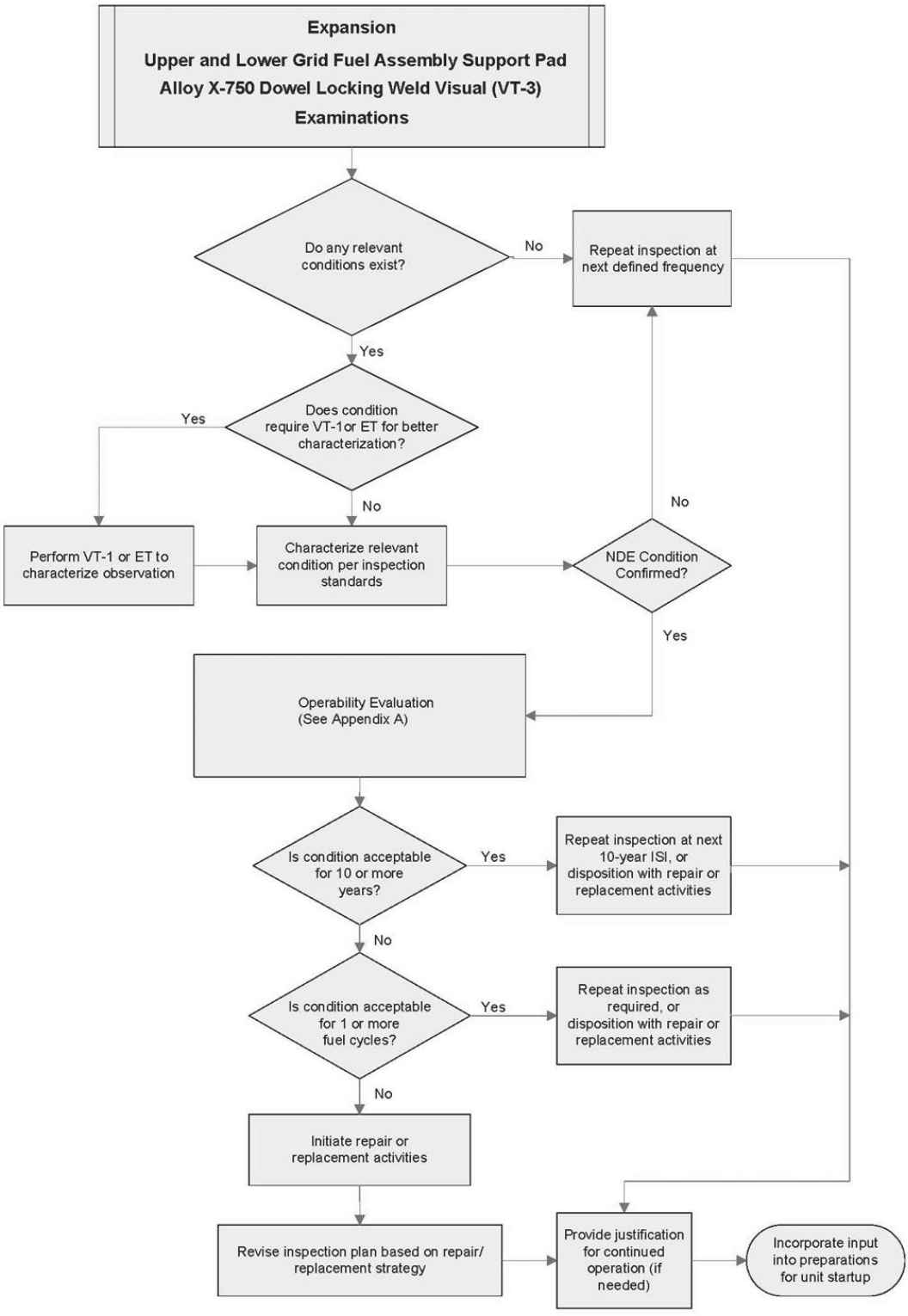
## B.2 EXPANSION COMPONENT ITEMS

Flowcharts for each of the Expansion component items are provided in this section. A separate sub-section is provided for each component item flowchart.



*** This record was final approved on 08/31/2023 13:45:20. (This statement was added by the PRIME system upon its validation)

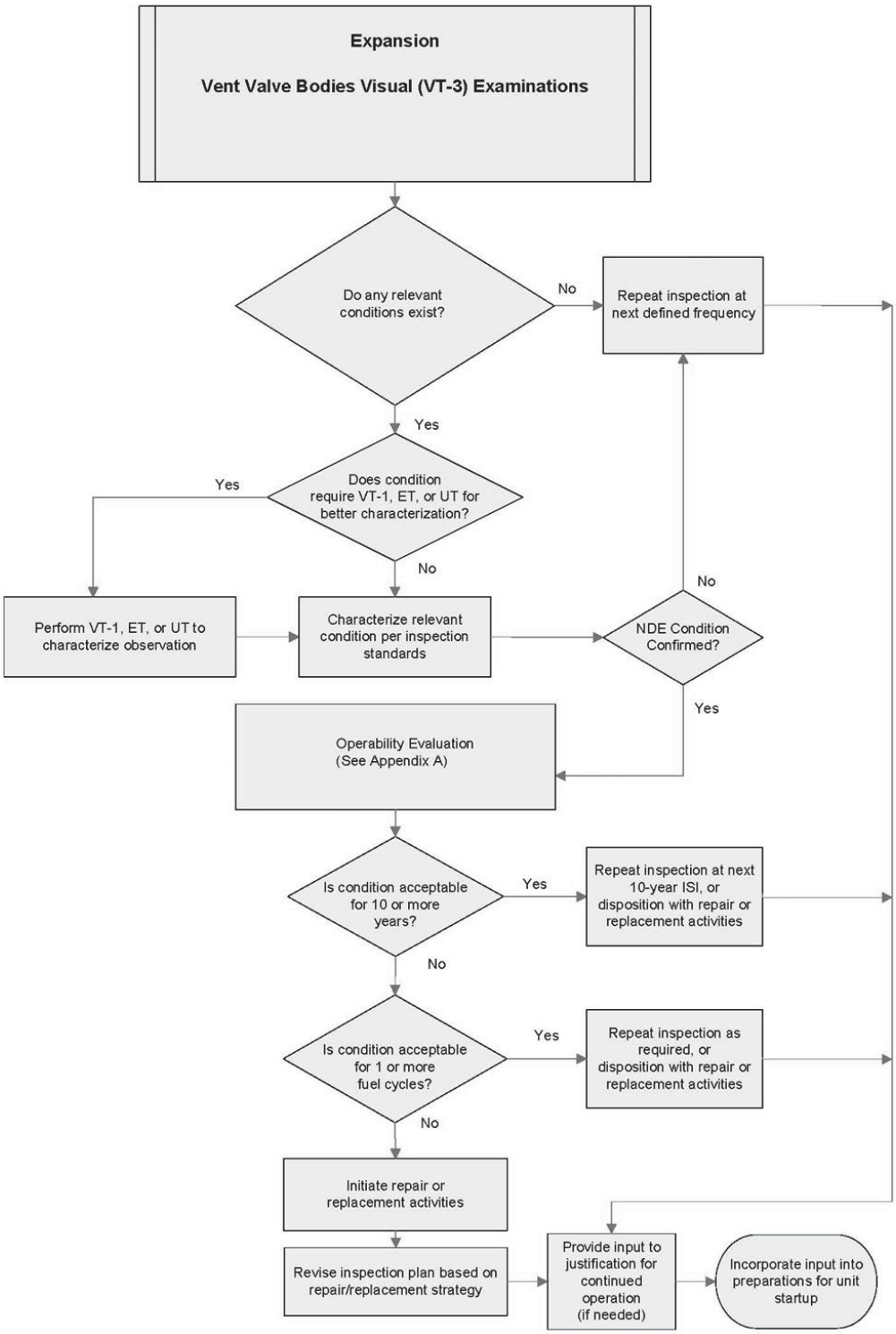
## B.2.1 Upper and Lower Grid Fuel Assembly Support Pad Dowel Welds






*** This record was final approved on 08/31/2023 13:45:20. (This statement was added by the PRIME system upon its validation)

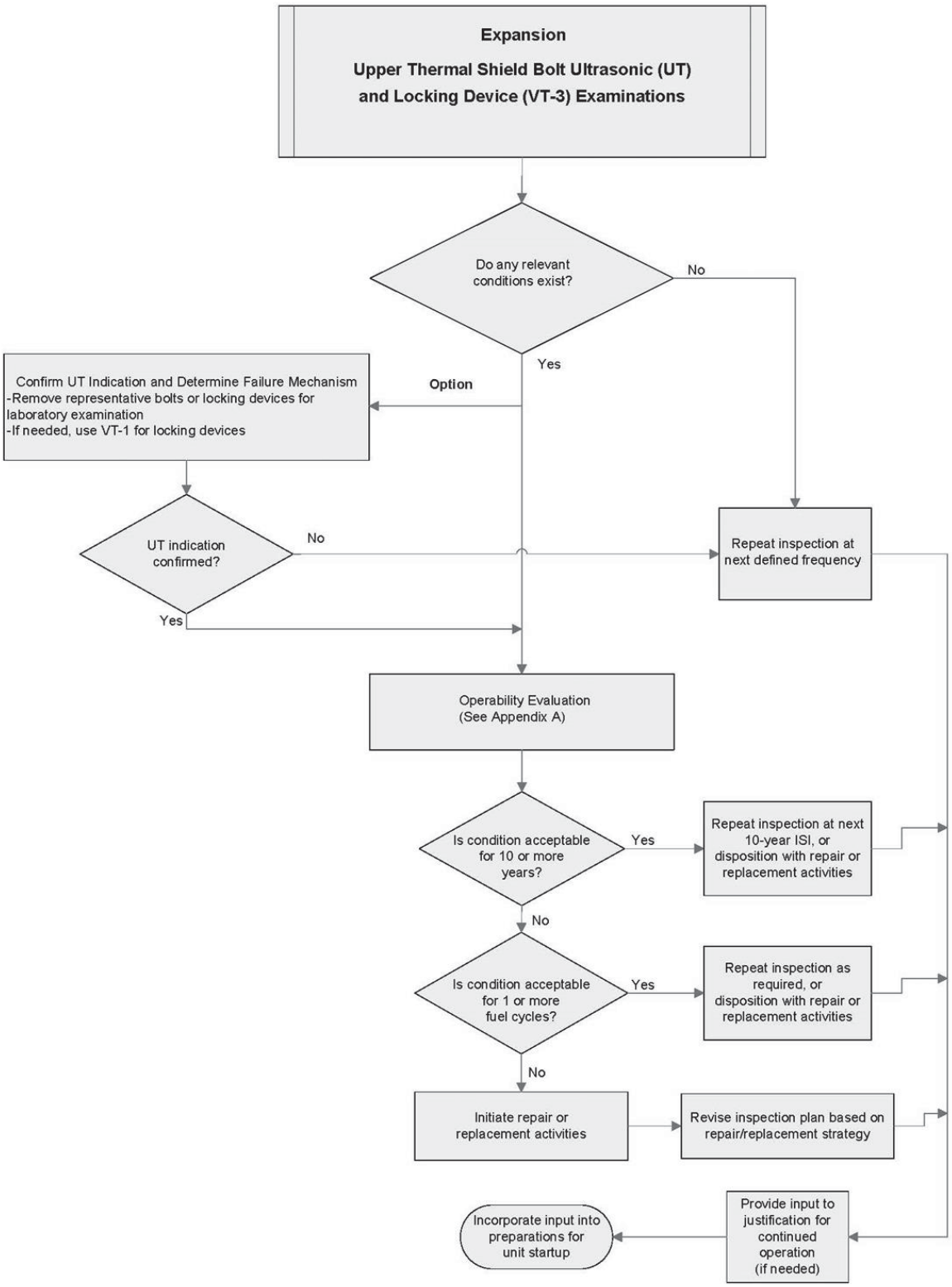
## B.2.2 Vent Valve Assembly Bodies






*** This record was final approved on 08/31/2023 13:45:20. (This statement was added by the PRIME system upon its validation)

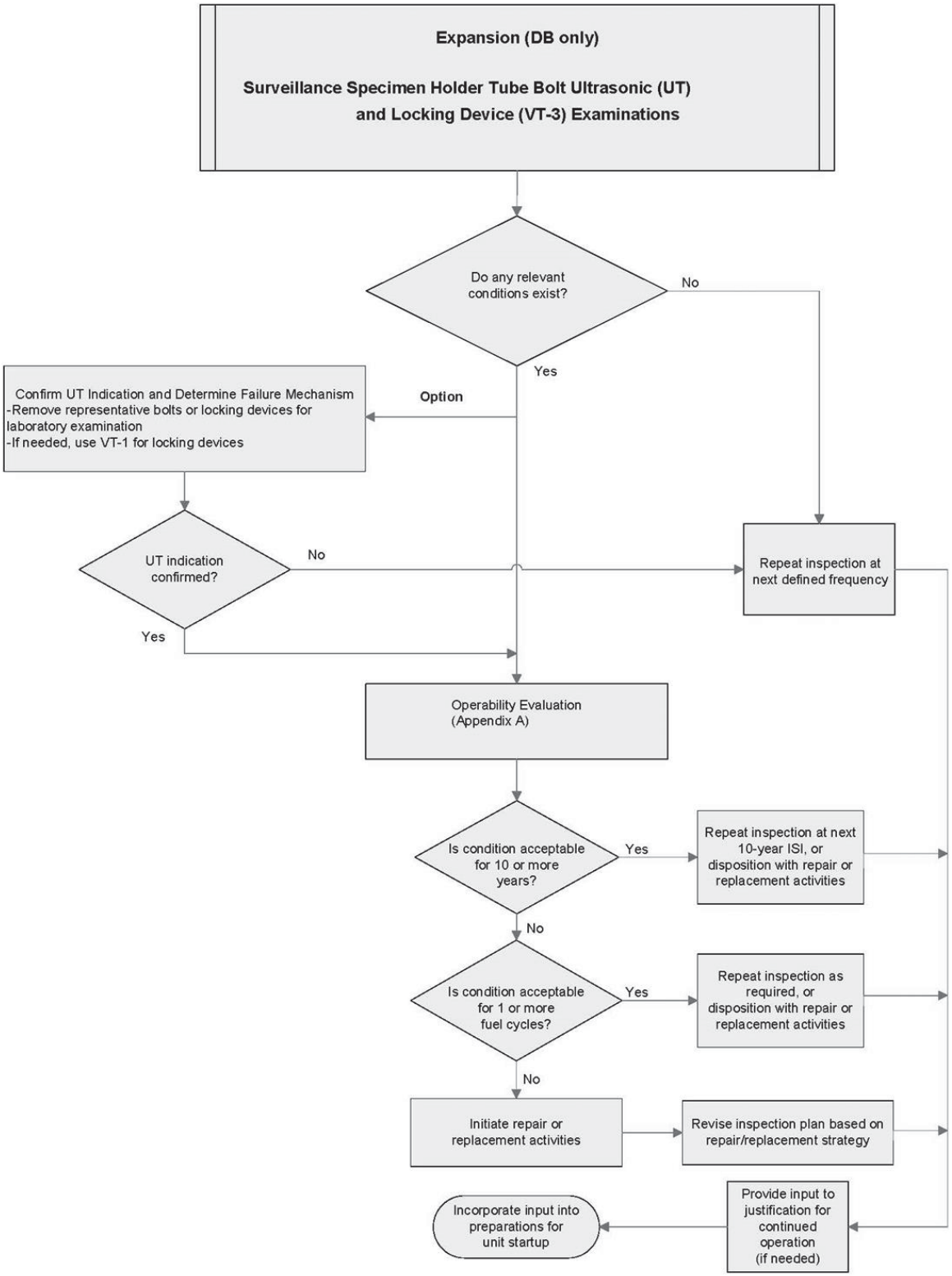
## B.2.3 Upper Thermal Shield Bolts and Locking Devices






*** This record was final approved on 08/31/2023 13:45:20. (This statement was added by the PRIME system upon its validation)

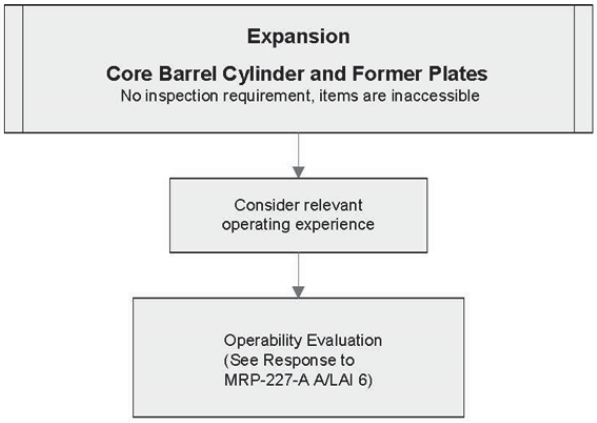
## B.2.4 Surveillance Specimen Holder Tube Bolts and Locking Devices






*** This record was final approved on 08/31/2023 13:45:20. (This statement was added by the PRIME system upon its validation)

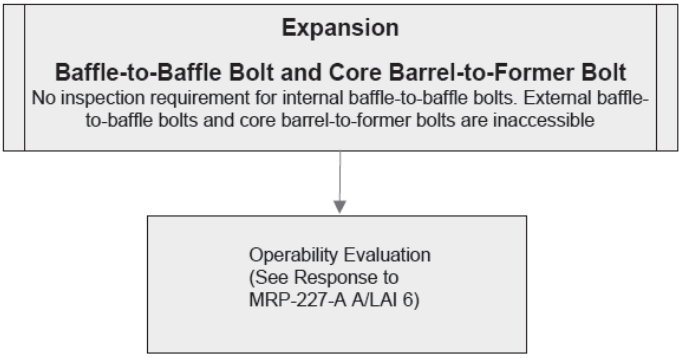
## B.2.5 Core Barrel Cylinder and Former Plates






*** This record was final approved on 08/31/2023 13:45:20. (This statement was added by the PRIME system upon its validation)

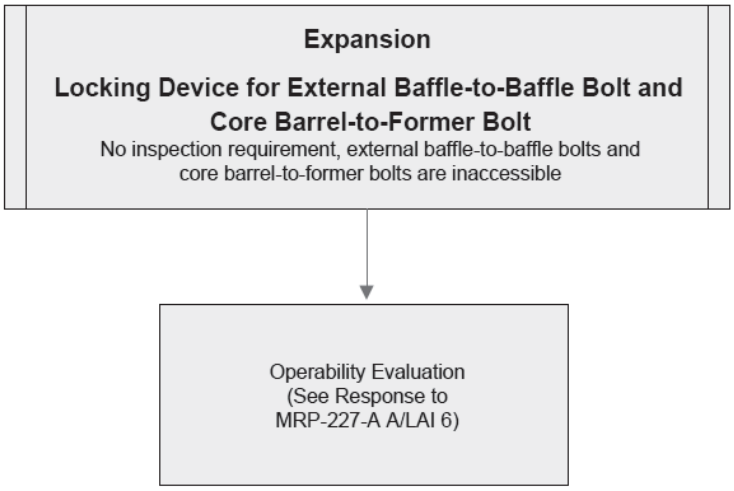
## B.2.6 Baffle-to-Baffle Bolts and Core Barrel-to-Former Bolts






*** This record was final approved on 08/31/2023 13:45:20. (This statement was added by the PRIME system upon its validation)

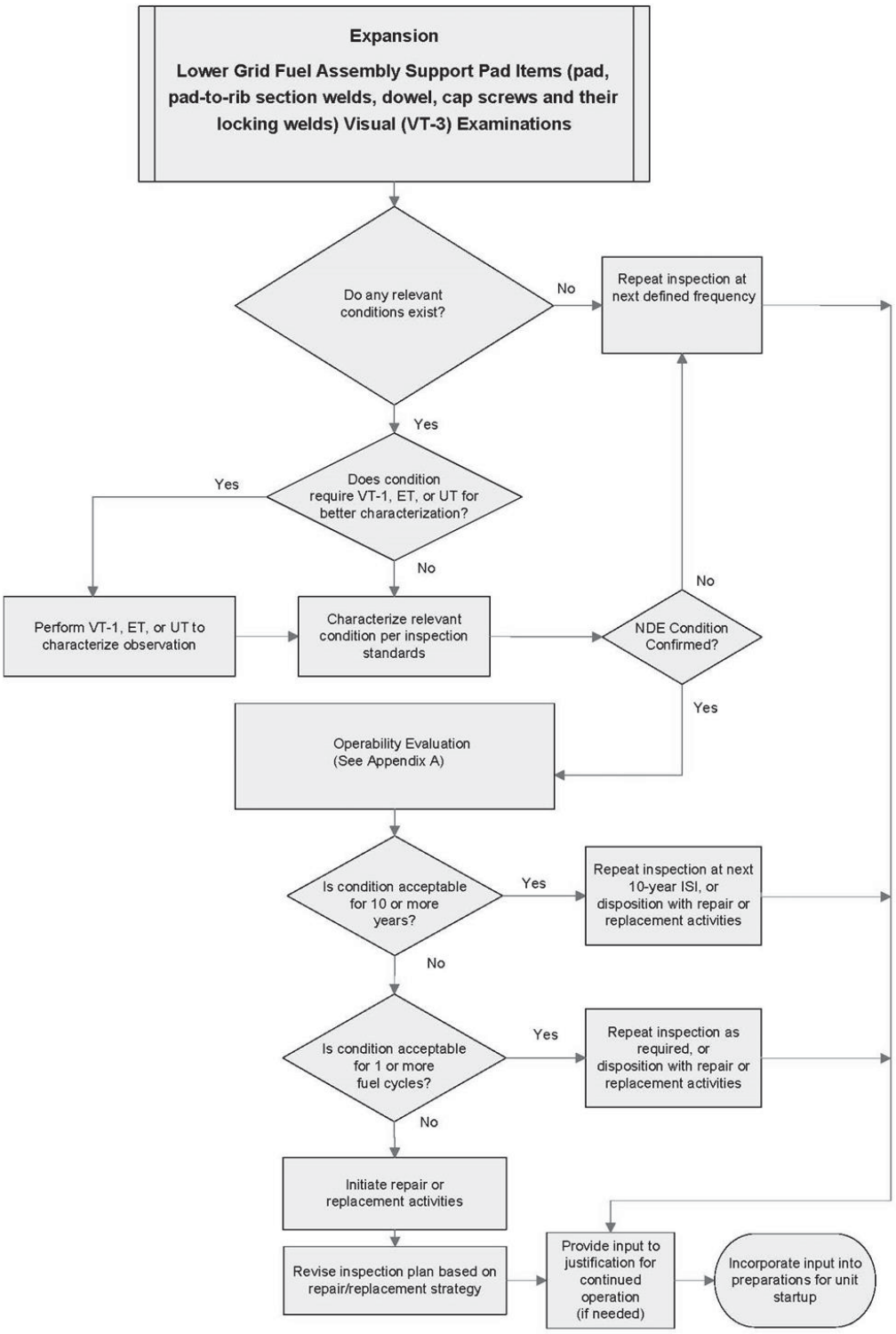
## B.2.7 Locking Devices for External Baffle-to-Baffle and Core Barrel-to-Former Bolts






*** This record was final approved on 08/31/2023 13:45:20. (This statement was added by the PRIME system upon its validation)

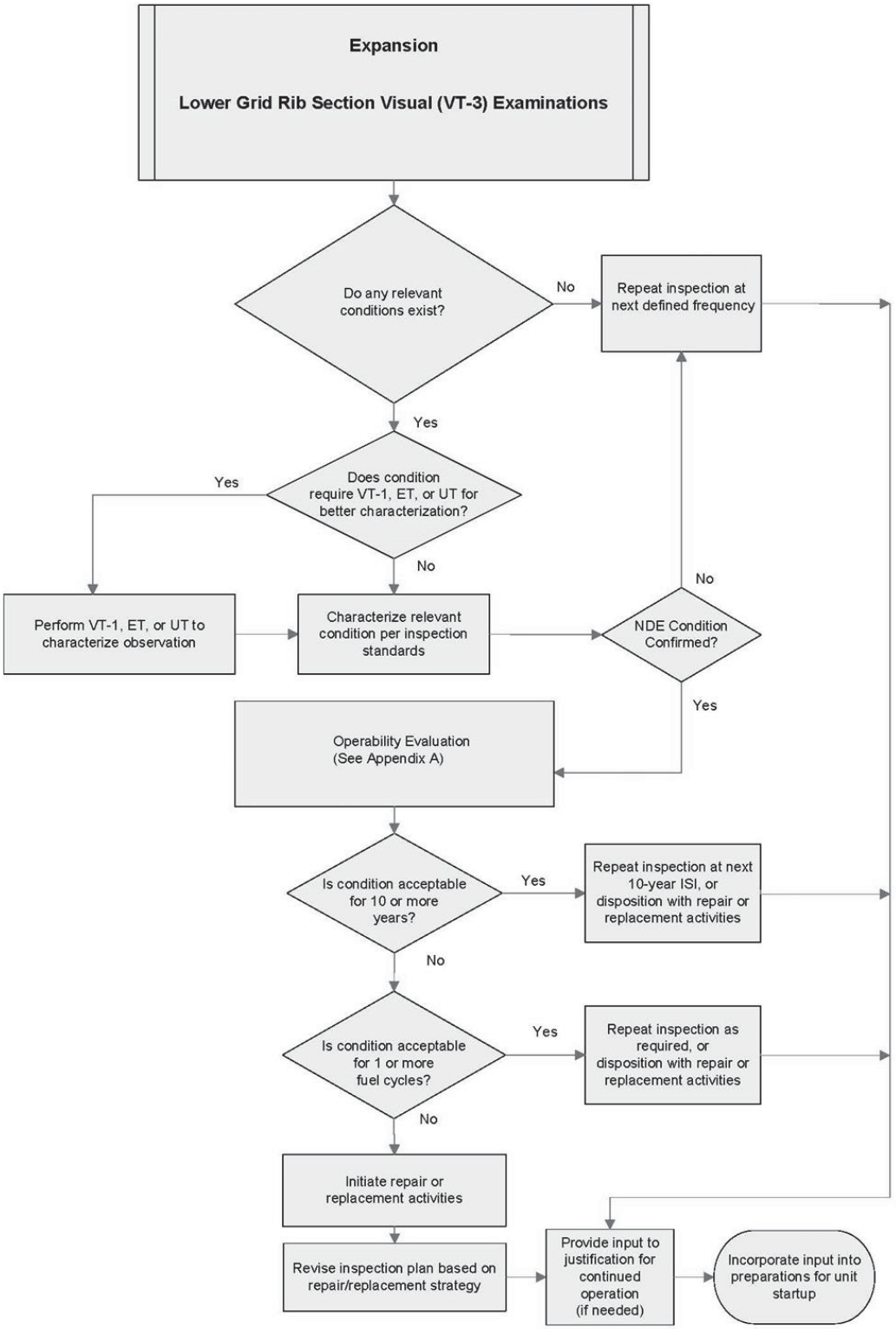
## B.2.8 Lower Grid Fuel Assembly Support Pad Items






*** This record was final approved on 08/31/2023 13:45:20. (This statement was added by the PRIME system upon its validation)

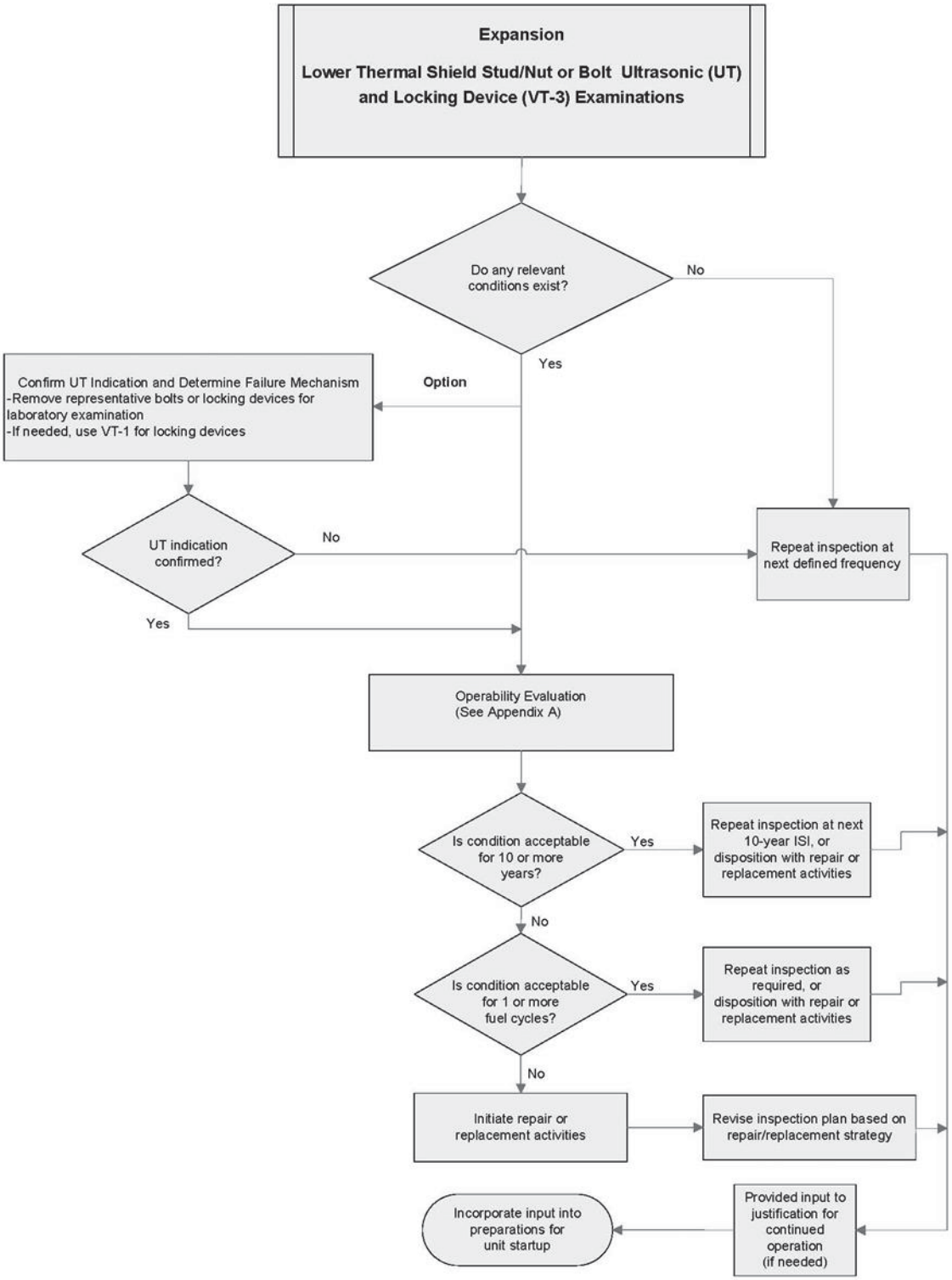
## B.2.9 Lower Grid Rib Section






*** This record was final approved on 08/31/2023 13:45:20. (This statement was added by the PRIME system upon its validation)

## B.2.10 Lower Thermal Shield Bolts and Locking Devices






*** This record was final approved on 08/31/2023 13:45:20. (This statement was added by the PRIME system upon its validation)

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

**CE Primary and Expansion Components**

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

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

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

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

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

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

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

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

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

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

CE-ID: 12 Lower Support Structure – Deep Beams



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| CE-ID: 2 | Core Shroud Assembly (Welded)<br>Core Shroud Plate-Former Plate Weld | | |
|---|---|---|---|
| Category: | Primary | Applicability: | Plant designs with core shrouds assembled in two vertical sections |
| Degradation Effect: | See MRP-227 Table 4-2 | | |
| Expansion Link: | See MRP-227 Table 4-2 | | |
| Function: | Direct coolant flow | | |

**Inspection**

| | |
|---|---|
| Method: | Visual examination. See MRP-227 Table 4-2. |
| Coverage: | See MRP-227 Table 4-2 |
| Observable Effect: | The specific relevant condition is a detectable crack-like surface indication. |

**Inputs and Assumptions**

There are several inputs and assumptions that are critical to the development of acceptance criteria for the core shroud welds. These items are stated below:

- The inspections identified in MRP-227 are intended to provide a sampling of potential locations of degradation. Under this approach, inspection of one side (surface) of the weld is intended to provide an adequate sampling for monitoring of IASCC.
- The change in resistance to fracture of the core shroud welds can be correlated to the accumulated fluence at each weld location. Welds that are subject to low fluence are considered to have a high degree of resistance to fracture. Correspondingly, those welds subject to high fluence have lower resistance to fracture. This region is expected to experience high fluence levels.
- It is assumed that the primary mode of crack growth to be expected in this region is irradiation assisted stress corrosion cracking.
- The prediction of crack growth is based on the stress intensity factor, K, calculated using linear elastic fracture mechanics. The rate of crack growth is dependent on the amount of neutron fluence that the weld is expected to accumulate over the licensed operating lifetime. Since there has been very limited experience of IASCC initiated cracks in this or comparable components or welds in operating PWRs to date, growth rate models must be obtained from other sources. One source is the models developed for predicting IASCC crack growth rate (CGR) in BWR components. The other source is the models developed based on laboratory data that may include test results for both PWR and BWR representative conditions. These models were considered to be appropriate for the prediction of crack growth due to IASCC in PWR internals.



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

**Core Shroud Plate-Former Plate Weld**

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

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

**Failure**

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

**Methodology**

| | |
|---|---|
| Goal: | Demonstrate that the cracking mechanism will not result in growth beyond the allowable crack length over the planned inspection interval. |
| Data Requirements: | 1. Surface crack length, as determined by visual inspection:<br>a. The 2013 edition of the ASME Code Section XI: IWA-3330(a) and Figure IWA-3330-1 provide requirements for the minimum allowable separation distance that will allow multiple cracks to be treated individually. This approach is part of the guidance in this report. The requirement in Section XI is that the ligament between adjacent cracks |



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

**Core Shroud Plate-Former Plate Weld**

must be greater than half of the thickness of the material. If this criterion is not met, the individual crack lengths and the length of the ligament between the cracks must be summed. This total length is then compared to the allowable length. Note that the 2007 and 2010 editions of ASME Code Section XI have the equivalent content to the 2013 edition and are also applicable.

b. The flaw proximity rules in BWRVIP-158 [62] may be optionally applied in lieu of those defined in ASME Section XI: IWA-3330(a) and Figure IWA-3330-1 when appropriate.

2. Flaw Depth
   a. For one-sided visual inspections, the flaw is assumed to be through-wall.
   b. Supplemental examinations may be used to determine depths of specific flaws as the basis for a flaw-specific evaluation, if needed.
3. Estimate of fast neutron fluence at crack location.
4. Steady-state normal operating stresses to be used to calculate IASCC crack growth rates.
   a. Axial stresses based on normal mechanical loads are required for horizontal flaws.
   b. Circumferential stresses based on normal mechanical loads are required for axial flaws.
   c. Stresses which have an insignificant net-through-wall value (average stress is near zero), such as weld residual stresses and thermal stresses due to local through-wall temperature gradients are considered to have minimal impact on the effective crack growth rates in through-wall flaws.
   d. Secondary weld residual and thermal stresses need to be considered in determination of through-wall crack growth rates in partial through-wall flaws, whose dimensions would have to be determined with supplemental UT examinations.
5. Limiting externally applied transient stresses to be used to calculate allowable flaw lengths.
6. More detailed load-deformation histories may be required for elastic-plastic or limit load calculations.

Analysis: All analyses require an assumption of the IASCC crack growth expected over the upcoming period of service. The methodology for analysis of a one-sided visual examination is based on an assumption of a through-wall flaw with weld residual and thermal stresses relieved. An appropriate crack growth rate model will be applied, which may be either K dependent or K independent. In order to apply the acceptance criteria to a period beyond one refueling cycle (not to exceed a 10-year re-inspection interval), follow-up action is required to verify the assumptions used in the predicted crack-growth rate. A re-inspection of the indication at a future specified outage, for example, would provide data that could be used to satisfy this verification requirement. Evaluate impact of fatigue



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

**Core Shroud Plate-Former Plate Weld**

crack growth on observed flaw. Failure of the welds is assumed to occur when unstable crack growth is initiated from the analyzed flaw. Three options are outlined for determining the limiting allowable flaw length, based on neutron dose. Analysis methods are suggested for both pre-inspection and generic analysis (Suggested Pre-Inspection Analysis) and for flaws observed in-service (Suggested Flaw Specific Analysis), where more detailed characteristics of the flaw and its location are known. In all cases, a more detailed evaluation may be completed using a semi-elliptic surface flaw, but such an evaluation would require more detailed inspection by UT.

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

Note:

1. Alternate fracture toughness values may be applied and justification for the alternate basis shall be submitted to the NRC for information within one year.
2. MRP-211 [66] provides the basis for the fracture toughness value.

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

Option 1: LEFM Analysis

a. Establish initial crack length (and depth if determined by supplemental examinations) based on inspection results.
b. The final crack dimensions are calculated by adding 10 years of crack growth under normal loading conditions.



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

**Core Shroud Plate-Former Plate Weld**

c. Ensure that the ΔK resulting from flow-induced vibration (FIV) is below the threshold for fatigue crack growth.

d. For primary loading at normal and upset loading conditions, the final stress intensity factor (K) for the flaw size from (b) must be lower than the fracture toughness for the material by a factor of at least 2.77. For the evaluation of partial through wall flaws, the stress intensity factor due to secondary loading does not require an additional safety factor and may be directly added to the adjusted stress intensity factor for primary loading (i.e., $2.77 \cdot K_P + K_S < K_{IC}$).

e. For the governing primary loading at emergency or faulted loading conditions, the final stress intensity factor (K) for the flaw size from (b) must be lower than the fracture toughness for the material by a factor of at least 1.39. For the evaluation of partial through wall flaws, the stress intensity factor due to secondary loading does not require an additional safety factor and may be directly added to the adjusted stress intensity factor for primary loading (i.e., $1.39 \cdot K_P + K_S < K_{IC}$).

Option 2: EPFM Analysis (for neutron dose levels > 0.5 dpa but < 5 dpa)

a. Establish initial crack length (and depth if determined by supplemental examinations) based on inspection results.

b. The final crack dimensions are calculated by adding 10 years of crack growth under normal loading conditions.

c. Ensure that the ΔK resulting from flow-induced vibration (FIV) is below the threshold for fatigue crack growth.

d. Develop the (J/T) material curve from the material J-R curve.

e. The $J_{applied}$ curve must include a safety factor of 2.77 for primary loading at normal and upset conditions, and a factor of 1.39 for primary loading at emergency and faulted conditions. For the evaluation of partial through wall flaws, the J curve due to secondary loading does not require an additional safety factor and may be directly added to the adjusted J curve for primary loading (i.e., $[2.77 \text{ or } 1.39] \cdot J_p + J_s < J_{applied}$).

f. The intersection of the material and applied (J/T) curves indicates the instability point. The flaw size at instability is determined from the $J_{applied}$ versus flaw size curve.

g. The flaw size at instability must be larger than the flaw size from (b), for the flaw to be acceptable.

The intersection of the material and applied (J/T) curves indicates the instability point. The load at instability is determined from the $J_{applied}$ versus load curve.

Option 3: Limit Load Analysis (For neutron fluence < $3 \times 10^{20}$ n/cm$^2$ at E>1 MeV or dose less than approximately 0.5 dpa) This region is expected to



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

**Core Shroud Plate-Former Plate Weld**

experience high fluence levels so it is unlikely that this option will be applicable.

a. Establish initial crack length (and depth if determined by supplemental examinations) based on inspection results.

b. The final crack dimensions are calculated by adding 10 years of crack growth under normal loading conditions.

c. Ensure that the ΔK resulting from flow-induced vibration (FIV) is below the threshold for fatigue crack growth.

d. Determine the bending moment (M) that can be tolerated as a function of the postulated flaw length.

e. The applied moment, increased by a factor of 1.39 (for primary loading at emergency and faulted conditions) or 2.77 (for primary loading at normal and upset conditions) must be less than the limit moment (from step d) for the flaw length determined in (b), for the flaw to be acceptable. For the evaluation of partial through wall flaws, the moment due to secondary loading does not require an additional safety factor and may be directly added to the adjusted moment for primary loading (i.e., $[2.77 \text{ or } 1.39] \cdot M_p + M_s < M_{applied}$).

Acceptance Criteria: The weld continues to perform its functional requirements with the projected flaw length at the end of the inspection interval.

Approach: Limit load, EPFM, or LEFM analysis

Additional Action(s): Note that the licensee shall submit the plant-specific fracture mechanics analysis justifying the detected flaws within one year after any inspection that triggers the expansion criteria as defined in Table 5-2 of MRP-227.



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| CE-ID: 2.1 | Core Shroud Assembly (Welded)<br>Remaining Axial Welds | | |
|---|---|---|---|
| Category: | Expansion | Applicability: | Plant designs with core shrouds assembled in two vertical sections |
| Degradation Effect: | See MRP-227 Table 4-5 | | |
| Primary Link: | See MRP-227 Table 4-5 | | |
| Function: | 1. Maintain core geometry.<br>2. Direct coolant flow. | | |

**Inspection**

| | |
|---|---|
| Method: | Visual examination. See MRP-227 Table 4-5 |
| Coverage: | See MRP-227 Table 4-5 |
| Observable Effect: | The specific relevant condition is a detectable crack-like surface indication. |

**Inputs and Assumptions**

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

- The inspections identified in MRP-227 are intended to provide a sampling of potential locations of degradation. Under this approach, inspection of one side (surface) of the weld is assumed to provide an adequate sampling for monitoring of Irradiation Assisted Stress Corrosion Cracking (IASCC).
- The change in resistance to fracture of the remaining axial welds can be correlated to the accumulated fluence at each weld location. Welds that are subject to low fluence are considered to have a high degree of resistance to fracture. Correspondingly, those welds subject to high fluence have lower resistance to fracture.
- It is assumed that the primary mode of crack growth to be expected in this region is stress corrosion cracking.
- The prediction of crack growth is based on the stress intensity factor, K, calculated using linear elastic fracture mechanics. The rate of crack growth is dependent on the amount of neutron fluence that the weld is expected to accumulate over the licensed operating lifetime. Since there has been very limited experience of IASCC initiated cracks in this or comparable components or welds in operating PWRs to date, growth rate models must be obtained from other sources. One source is the models developed for predicting IASCC crack growth rate (CGR) in BWR components. The other source is the models developed based on laboratory data that may include test results for both PWR and BWR representative conditions. These models were considered to be appropriate for the prediction of crack growth due to IASCC in PWR internals.
    - For weld locations subjected to fluence levels less than or equal to $5x10^{20}$ n/cm$^2$ (E>1MeV)(0.75 dpa), the maximum of the crack growth rate in 2017 ASME Section XI Appendix C-8520 **or** the



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**CE-ID: 2.1** **Core Shroud Assembly (Welded)**

**Remaining Axial Welds**

crack growth equation in ASME Code Case N-889 [52] will be used. The conditions for the use of Code Case N-889 identified in Regulatory Guide RG 1.147, Rev. 20 [67] must be met.

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

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

**Failure**

| | |
|---|---|
| Failure Mechanism: | Cracking (IASCC) |
| Failure Effect: | 1. Core damage caused by event – require maintenance of coolable geometry.<br>2. Damage to peripheral fuel assemblies.<br>3. Through-wall crack provides leak path through shroud. |
| Failure Criteria: | Actively growing through-wall flaws may require mitigation.<br>An existing flaw is unacceptable if the flaw length projected at the next inspection cycle is beyond the allowable crack length. |

**Methodology**

| | |
|---|---|
| Goal: | Demonstrate that the cracking mechanism will not result in growth beyond the allowable crack length over the planned inspection interval. |
| Data Requirements: | 1. Surface crack length as determined by visual inspection<br>a. The 2013 edition of the ASME Code Section XI: IWA-3330(a) and Figure IWA-3330-1 provide requirements for the minimum allowable separation distance that will serve as the basis for guidance in this report. The requirement in Section XI is that the ligament between adjacent cracks must be greater than half of the thickness of the material. If this criterion is not met, the individual |



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**CE-ID: 2.1** **Core Shroud Assembly (Welded)**

**Remaining Axial Welds**

crack lengths and the length of the ligament between the cracks must be summed. This total length is then compared to the allowable length. Note that the 2007 and 2010 editions of ASME Code Section XI have the equivalent content to the 2013 edition and are also applicable.

b. The flaw proximity rules in BWRVIP-158 [62] may be optionally applied in lieu of those defined in ASME Section XI: IWA-3330(a) and Figure IWA-3330-1 when appropriate.

2. Flaw Depth
   a. For one-sided visual inspections, the flaw is assumed to be through-wall.
   b. Supplemental examinations may be used to determine flaw depth, for a flaw-specific criterion, if needed.
3. Estimate of fast neutron fluence at crack location.
4. Steady-state normal operating stresses to be used to calculate IASCC crack growth rates.
   a. Horizontal stresses based on normal mechanical loads are required for axial flaws.
   b. Stresses which have an insignificant net-through-wall value (average stress is near zero), such as weld residual stresses and thermal stresses due to local through-wall temperature gradients are considered to have minimal impact on the effective crack growth rates in through-wall flaws.
   c. Secondary weld residual and thermal stresses need to be considered in determination of circumferential and through-wall crack growth rates in partial through-wall flaws, whose dimensions would have to be determined with supplemental UT examinations.
5. Limiting externally applied transient stresses to be used to calculate allowable flaw lengths. For example, stresses arising from pressure, mechanical and thermal loads would be included in this calculation,
   a. More detailed load-deformation histories may be required for elastic-plastic or limit load calculations, if these calculations are necessary.

Analysis: All analyses require an assumption of the IASCC crack growth expected over the upcoming period of service. The methodology for analysis of a one-sided visual examination is based on an assumption of a through-wall flaw with weld residual and thermal stresses relieved. An appropriate crack growth rate model will be applied, which may be either K dependent or K independent. In order to apply the acceptance criteria to a period beyond one refueling cycle (not to exceed a 10-year re-inspection interval), follow-up action is required to verify the assumptions used in the predicted crack-growth rate. A re-inspection of the indication at a future specified outage, for example, would provide data that could be used to satisfy this verification requirement. Evaluate impact of fatigue



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## CE-ID: 2.1 Core Shroud Assembly (Welded)

### Remaining Axial Welds

crack growth on observed flaw. Failure of the weld is assumed to occur when unstable crack growth is initiated from the analyzed flaw. Three options are outlined for determining the limiting allowable flaw length, based on neutron dose. Analysis methods are suggested for both pre-inspection and generic analysis (Suggested Pre-Inspection Analysis) and for flaws observed in-service (Suggested Flaw Specific Analysis), where more detailed characteristics of the flaw and its location are known. In all cases, a more detailed evaluation may be completed using a semi-elliptic surface flaw, but such an evaluation would require more detailed inspection by UT.

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

Note:

1. Alternate fracture toughness values may be applied and justification for the alternate basis shall be submitted to the NRC for information within one year.
2. MRP-211 [66] provides the basis for the fracture toughness value.

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



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**CE-ID: 2.1** **Core Shroud Assembly (Welded)**

**Remaining Axial Welds**

Option 1: LEFM Analysis

a. Establish initial crack length (and depth if determined by supplemental examinations) based on inspection results.
b. The final crack dimensions are calculated by adding 10 years of crack growth under normal loading conditions.
c. Ensure that the ΔK resulting from flow-induced vibration (FIV) is below the threshold for fatigue crack growth.
d. For primary loading at normal and upset loading conditions, the final stress intensity factor (K) for the flaw size from (b) must be lower than the fracture toughness for the material by a factor of at least 2.77. For the evaluation of partial through wall flaws, the stress intensity factor due to secondary loading does not require an additional safety factor and may be directly added to the adjusted stress intensity factor for primary loading (i.e., $2.77 \cdot K_P + K_S < K_{IC}$).
e. For the governing primary loading at emergency or faulted loading conditions, the final stress intensity factor (K) for the flaw size from (b) must be lower than the fracture toughness for the material by a factor of at least 1.39. For the evaluation of partial through wall flaws, the stress intensity factor due to secondary loading does not require an additional safety factor and may be directly added to the adjusted stress intensity factor for primary loading (i.e., $1.39 \cdot K_P + K_S < K_{IC}$).

Option 2: EPFM Analysis (for neutron dose levels > 0.5 dpa but < 5 dpa)

a. Establish initial crack length (and depth if determined by supplemental examinations) based on inspection results.
b. The final crack dimensions are calculated by adding 10 years of crack growth under normal loading conditions.
c. Ensure that the ΔK resulting from flow-induced vibration (FIV) is below the threshold for fatigue crack growth.
d. Develop the (J/T) material curve from the material J-R curve.
e. The $J_{applied}$ curve must include a safety factor of 2.77 for primary loading at normal and upset conditions, and a factor of 1.39 for primary loading at emergency and faulted conditions. For the evaluation of partial through wall flaws, the J curve due to secondary loading does not require an additional safety factor and may be directly added to the adjusted J curve for primary loading (i.e., $[2.77 \text{ or } 1.39] \cdot J_p + J_s < J_{applied}$).
f. The intersection of the material and applied (J/T) curves indicates the instability point. The flaw size at instability is determined from the $J_{applied}$ versus flaw size curve.
g. The flaw size at instability must be larger than the flaw size from (b), for the flaw to be acceptable.



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**CE-ID: 2.1** **Core Shroud Assembly (Welded)**

**Remaining Axial Welds**

The intersection of the material and applied (J/T) curves indicates the instability point. The load at instability is determined from the $J_{applied}$ versus load curve.

Option 3: Limit Load Analysis (For neutron fluence < $3x10^{20}$ n/cm$^2$ at E>1 MeV or dose less than approximately 0.5 dpa)

a. Establish initial crack length (and depth if determined by supplemental examinations) based on inspection results.

b. The final crack dimensions are calculated by adding 10 years of crack growth under normal loading conditions.

c. Ensure that the ΔK resulting from flow-induced vibration (FIV) is below the threshold for fatigue crack growth.

d. Determine the bending moment (M) that can be tolerated as a function of the postulated flaw length.

e. The applied moment, increased by a factor of 1.39 (for primary loading at emergency and faulted conditions) or 2.77 (for primary loading at normal and upset conditions) must be less than the limit moment (from step d) for the flaw length determined in (b), for the flaw to be acceptable. For the evaluation of partial through wall flaws, the moment due to secondary loading does not require an additional safety factor and may be directly added to the adjusted moment for primary loading (i.e., $[2.77 \text{ or } 1.39]\cdot M_p + M_s < M_{applied}$).

Acceptance Criteria: The weld continues to perform its functional requirements with the projected flaw length at the end of the inspection interval.

Approach: Limit load, EPFM, or LEFM analysis

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



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| CE-ID: 3 | Core Shroud Assembly (Welded)<br>Shroud Plates | | |
|---|---|---|---|
| Category: | Primary | Applicability: | Plant designs with core shrouds assembled with full-height shroud plates |
| Degradation Effect: | See MRP-227 Table 4-2 | | |
| Expansion Link: | See MRP-227 Table 4-2 | | |
| Function: | 1. Maintain core geometry.<br>2. Direct coolant flow. | | |

**Inspection**

| | |
|---|---|
| Method: | Visual examination. See MRP-227 Table 4-2. |
| Coverage: | See MRP-227 Table 4-2 |
| Observable Effect: | The specific relevant condition is a detectable crack-like surface indication. |

**Inputs and Assumptions**

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

- The inspections identified in MRP-227 are intended to provide a sampling of potential locations of degradation. Under this approach, inspection of one side (surface) of the weld is assumed to provide an adequate sampling for monitoring of Irradiation Assisted Stress Corrosion Cracking (IASCC).
- The change in resistance to fracture of the shroud plates welds can be correlated to the accumulated fluence at each weld location. Welds that are subject to low fluence are considered to have a high degree of resistance to fracture. Correspondingly, those welds subject to high fluence have lower resistance to fracture.
- It is assumed that the primary mode of crack growth to be expected in this region is stress corrosion cracking.
- The prediction of crack growth is based on the stress intensity factor, K, calculated using linear elastic fracture mechanics. The rate of crack growth is dependent on the amount of neutron fluence that the weld is expected to accumulate over the licensed operating lifetime. Since there has been very limited experience of IASCC initiated cracks in this or comparable components or welds in operating PWRs to date, growth rate models must be obtained from other sources. One source is the models developed for predicting IASCC crack growth rate (CGR) in BWR components. The other source is the models developed based on laboratory data that may include test results for both PWR and BWR representative conditions. These models were considered to be appropriate for the prediction of crack growth due to IASCC in PWR internals.
- For weld locations subjected to fluence levels less than or equal to $5x10^{20}$ n/cm$^2$ (E>1MeV)(0.75 dpa), the maximum of the crack growth rate in 2017 ASME



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**CE-ID: 3** **Core Shroud Assembly (Welded)**

**Shroud Plates**

Section XI Appendix C-8520 **or** the crack growth equation in ASME Code Case N-889 [52] will be used. The conditions for the use of Code Case N-889 identified in Regulatory Guide RG 1.147, Rev. 20 [67] must be met.

- For weld locations subjected to fluence levels greater than $5x10^{20}$ n/cm$^2$ (E>1MeV) (0.75 dpa) up to and including $1.4x10^{22}$ n/cm$^2$ (E>1MeV)(20 dpa), the crack growth equation in ASME Code Case N-889 [52] must be used. The conditions for the use of Code Case N-889 identified in Regulatory Guide RG 1.147, Rev. 20 [67] must be met for the development of acceptance criteria.
- For weld locations subjected to fluence levels greater than $1.4x10^{22}$ n/cm$^2$ (E>1MeV)(20 dpa), the CGR model in EPRI Topical Report, *Models of Irradiation-Assisted Stress Corrosion Cracking of Austenitic Stainless Steels in Light Water Reactor Environments, Volume 2. Disposition Curves Application* [46], will be used. The acceptance criteria developed for fluence levels above $1.4x10^{22}$ n/cm$^2$ (E>1 MeV)(20 dpa) must be approved by the NRC as discussed in Reg. Guide 1.147 Rev. 20 [67].
- Depending on the magnitude of cyclic loads, such as thermal transients, additional crack growth as a result of these loads may need to be considered.
- Acceptance criteria can be developed for the entire operating license of a given plant by using predicted end-of-license fluence values. However, if there are changes to these fluence projections, such as in the event of a power uprate or change in core loading pattern, it would be necessary to confirm that the inputs selected based on fluence, such as IASCC growth rate and fracture toughness, remain applicable until the end of the operating license.

**Failure**

| | |
|---|---|
| Failure Mechanism: | Cracking (IASCC) |
| Failure Effect: | 1. Core damage caused by event – require maintenance of coolable geometry.<br>2. Damage to peripheral fuel assemblies.<br>3. Through-wall crack provides leak path through shroud. |
| Failure Criteria: | Actively growing through-wall flaws may require mitigation.<br>An existing flaw is unacceptable if the flaw length projected at the next inspection cycle is beyond the allowable crack length. |

**Methodology**

| | |
|---|---|
| Goal: | Demonstrate that the cracking mechanism will not result in growth beyond the allowable crack length over the planned inspection interval. |
| Data Requirements: | 1. Surface crack length as determined by visual inspection<br>a. The 2013 edition of the ASME Code Section XI: IWA-3330(a) and Figure IWA-3330-1 provide requirements for the minimum allowable separation distance that will serve as the basis for guidance in this report. The requirement in Section XI is that the ligament between adjacent cracks must be greater than half of the |



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**CE-ID: 3** **Core Shroud Assembly (Welded)**

**Shroud Plates**

thickness of the material. If this criterion is not met, the individual crack lengths and the length of the ligament between the cracks must be summed. This total length is then compared to the allowable length. Note that the 2007 and 2010 editions of ASME Code Section XI have the equivalent content to the 2013 edition and are also applicable.

b. The flaw proximity rules in BWRVIP-158 [62] may be optionally applied in lieu of those defined in ASME Section XI: IWA-3330(a) and Figure IWA-3330-1 when appropriate.

2. Flaw Depth
   a. For one-sided visual inspections, the flaw is assumed to be through-wall.
   b. Supplemental examinations may be used to determine flaw depth, for a flaw-specific criterion, if needed.
3. Estimate of fast neutron fluence at crack location.
4. Steady-state normal operating stresses to be used to calculate IASCC crack growth rates.
   a. Vertical stresses based on normal mechanical loads are required for horizontal flaws.
   b. Horizontal stresses based on normal mechanical loads are required for vertical flaws.
   c. Stresses which have an insignificant net-through-wall value (average stress is near zero), such as weld residual stresses and thermal stresses due to local through-wall temperature gradients are considered to have minimal impact on the effective crack growth rates in through-wall flaws.
   d. Secondary weld residual and thermal stresses need to be considered in determination of circumferential and through-wall crack growth rates in partial through-wall flaws, whose dimensions would have to be determined with supplemental UT examinations.
5. Limiting externally applied transient stresses to be used to calculate allowable flaw lengths. For example, stresses arising from pressure, mechanical and thermal loads would be included in this calculation,
   a. More detailed load-deformation histories may be required for elastic-plastic or limit load calculations, if these calculations are necessary.

Analysis: All analyses require an assumption of the IASCC crack growth expected over the upcoming period of service. The methodology for analysis of a one-sided visual examination is based on an assumption of a through-wall flaw with weld residual and thermal stresses relieved. An appropriate crack growth rate model will be applied, which may be either K dependent or K independent. In order to apply the acceptance criteria to a period beyond one refueling cycle (not to exceed a 10-year re-inspection interval), follow-up action is required to verify the



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**CE-ID: 3** **Core Shroud Assembly (Welded)**

## Shroud Plates

assumptions used in the predicted crack-growth rate. A re-inspection of the indication at a future specified outage, for example, would provide data that could be used to satisfy this verification requirement.

Failure of the shroud plate weld is assumed to occur when unstable crack growth is initiated from the analyzed flaw. Three options are outlined for determining the limiting allowable flaw length, based on neutron dose. Analysis methods are suggested for both pre-inspection and generic analysis (Suggested Pre-Inspection Analysis) and for flaws observed in-service (Suggested Flaw Specific Analysis), where more detailed characteristics of the flaw and its location are known. In all cases, a more detailed evaluation may be completed using a semi-elliptic surface flaw, but such an evaluation would require more detailed inspection by UT.

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

Note:

1. Alternate fracture toughness values may be applied and justification for the alternate basis shall be submitted to the NRC for information within one year.
2. MRP-211 [66] provides the basis for the fracture toughness value.

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

Option 1: LEFM Analysis



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**CE-ID: 3** **Core Shroud Assembly (Welded)**

**Shroud Plates**

a. Establish initial crack length (and depth if determined by supplemental examinations) based on inspection results.

b. The final crack dimensions are calculated by adding 10 years of crack growth under normal loading conditions.

c. Ensure that the ΔK resulting from flow-induced vibration (FIV) is below the threshold for fatigue crack growth.

d. For primary loading at normal and upset loading conditions, the final stress intensity factor (K) for the flaw size from (b) must be lower than the fracture toughness for the material by a factor of at least 2.77. For the evaluation of partial through wall flaws, the stress intensity factor due to secondary loading does not require an additional safety factor and may be directly added to the adjusted stress intensity factor for primary loading (i.e., $2.77 \cdot K_P + K_S < K_{IC}$).

e. For the governing primary loading at emergency or faulted loading conditions, the final stress intensity factor (K) for the flaw size from (b) must be lower than the fracture toughness for the material by a factor of at least 1.39. For the evaluation of partial through wall flaws, the stress intensity factor due to secondary loading does not require an additional safety factor and may be directly added to the adjusted stress intensity factor for primary loading (i.e., $1.39 \cdot K_P + K_S < K_{IC}$).

Option 2: EPFM Analysis (for neutron dose levels > 0.5 dpa but < 5 dpa)

a. Establish initial crack length (and depth if determined by supplemental examinations) based on inspection results.

b. The final crack dimensions are calculated by adding 10 years of crack growth under normal loading conditions.

c. Ensure that the ΔK resulting from flow-induced vibration (FIV) is below the threshold for fatigue crack growth.

d. Develop the (J/T) material curve from the material J-R curve.

e. The $J_{applied}$ curve must include a safety factor of 2.77 for primary loading at normal and upset conditions, and a factor of 1.39 for primary loading at emergency and faulted conditions. For the evaluation of partial through wall flaws, the J curve due to secondary loading does not require an additional safety factor and may be directly added to the adjusted J curve for primary loading (i.e., $[2.77 \text{ or } 1.39] \cdot J_p + J_s < J_{applied}$).

f. The intersection of the material and applied (J/T) curves indicates the instability point. The flaw size at instability is determined from the $J_{applied}$ versus flaw size curve.

g. The flaw size at instability must be larger than the flaw size from (b), for the flaw to be acceptable.

The intersection of the material and applied (J/T) curves indicates the instability point. The load at instability is determined from the Japplied versus load curve.



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**CE-ID: 3** **Core Shroud Assembly (Welded)**

**Shroud Plates**

Option 3: Limit Load Analysis (For neutron fluence < $3x10^{20}$ n/cm$^2$ at E>1 MeV or dose less than approximately 0.5 dpa)

a. Establish initial crack length (and depth if determined by supplemental examinations) based on inspection results.

b. The final crack dimensions are calculated by adding 10 years of crack growth under normal loading conditions.

c. Ensure that the $\Delta K$ resulting from flow-induced vibration (FIV) is below the threshold for fatigue crack growth.

d. Determine the bending moment (M) that can be tolerated as a function of the postulated flaw length.

e. The applied moment, increased by a factor of 1.39 (for primary loading at emergency and faulted conditions) or 2.77 (for primary loading at normal and upset conditions) must be less than the limit moment (from step d) for the flaw length determined in (b), for the flaw to be acceptable. For the evaluation of partial through wall flaws, the moment due to secondary loading does not require an additional safety factor and may be directly added to the adjusted moment for primary loading (i.e., $[2.77 \text{ or } 1.39] \cdot M_p + M_s < M_{applied}$).

| | |
|---|---|
| Acceptance Criteria: | The shroud plate welds continues to perform its functional requirements with the projected flaw length at the end of the inspection interval. |
| Approach: | Limit load, EPFM, or LEFM analysis |
| Additional Action(s): | Note that the licensee shall submit the plant-specific fracture mechanics analysis justifying the detected flaws within one year after any inspection that triggers the expansion criteria as defined in Table 5-2 of MRP-227. |



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| CE-ID: 3.1 | Core Shroud Assembly (Welded)<br>Remaining Axial Welds | | |
|---|---|---|---|
| Category: | Expansion | Applicability: | Plant designs with core shrouds assembled with full-height shroud plates |
| Degradation Effect: | See MRP-227 Table 4-5 | | |
| Primary Link: | See MRP-227 Table 4-5 | | |
| Function: | Maintain dimensional stability of core shroud plus ribs and rings | | |

**Inspection**

| | |
|---|---|
| Method: | Visual examination. See MRP-227 Table 4-5. |
| Coverage: | See MRP-227 Table 4-5 |
| Observable Effect: | The specific relevant condition is a detectable crack-like surface indication. |

**Inputs and Assumptions**

There are several inputs and assumptions that are critical to the development of acceptance criteria for the remaining axial weld locations. These items are stated below:

- The inspections identified in MRP-227 are intended to provide a sampling of potential locations of degradation. Under this approach, inspection of one side (surface) of the welds is assumed to provide an adequate sampling for monitoring of Irradiation Assisted Stress Corrosion Cracking (IASCC).
- The change in resistance to fracture of the remaining axial welds can be correlated to the accumulated fluence at each weld location. Welds that are subject to low fluence are considered to have a high degree of resistance to fracture. Correspondingly, those welds subject to high fluence have lower resistance to fracture. The remaining axial welds are expected to be highly irradiated.
- It is assumed that the primary mode of crack growth to be expected in this region is stress corrosion cracking.
- The prediction of crack growth is based on the stress intensity factor, K, calculated using linear elastic fracture mechanics. The rate of crack growth is dependent on the amount of neutron fluence that the weld is expected to accumulate over the licensed operating lifetime. Since there has been very limited experience of IASCC initiated cracks in this or comparable components or welds in operating PWRs to date, growth rate models must be obtained from other sources. One source is the models developed for predicting IASCC crack growth rate (CGR) in BWR components. The other source is the models developed based on laboratory data that may include test results for both PWR and BWR representative conditions. These models were considered to be appropriate for the prediction of crack growth due to IASCC in PWR internals.
    - For weld locations subjected to fluence levels less than or equal to $5x10^{20}$ n/cm$^2$ (E>1MeV)(0.75 dpa), the maximum of the



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**CE-ID: 3.1** **Core Shroud Assembly (Welded)**

**Remaining Axial Welds**

crack growth rate in 2017 ASME Section XI Appendix C-8520 **or** the crack growth equation in ASME Code Case N-889 [52] will be used. The conditions for the use of Code Case N-889 identified in Regulatory Guide RG 1.147, Rev. 20 [67] must be met.

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

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

**Failure**

| | |
|---|---|
| Failure Mechanism: | Cracking (IASCC) |
| Failure Effect: | 1. Core damage caused by event – require maintenance of coolable geometry.<br>2. Damage to peripheral fuel assemblies.<br>3. Through-wall crack provides leak path through shroud. |
| Failure Criteria: | Actively growing through-wall flaws may require mitigation.<br>An existing flaw is unacceptable if the flaw length projected at the next inspection cycle is beyond the allowable crack length.<br>No observable damage in corresponding sections of peripheral fuel assemblies. |

**Methodology**



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**CE-ID: 3.1** **Core Shroud Assembly (Welded)**

**Remaining Axial Welds**

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

Data Requirements:

1. Surface crack length as determined by visual inspection
    a. The 2013 edition of the ASME Code Section XI: IWA-3330(a) and Figure IWA-3330-1 provide requirements for the minimum allowable separation distance that will serve as the basis for guidance in this report. The requirement in Section XI is that the ligament between adjacent cracks must be greater than half of the thickness of the material. If this criterion is not met, the individual crack lengths and the length of the ligament between the cracks must be summed. This total length is then compared to the allowable length. Note that the 2007 and 2010 editions of ASME Code Section XI have the equivalent content to the 2013 edition and are also applicable.
    b. The flaw proximity rules in BWRVIP-158 [62] may be optionally applied in lieu of those defined in ASME Section XI: IWA-3330(a) and Figure IWA-3330-1 when appropriate.
2. Flaw Depth
    a. For one-sided visual inspections, the flaw is assumed to be through-wall.
    b. Supplemental examinations may be used to determine flaw depth, for a flaw-specific criterion, if needed.
3. Estimate of fast neutron fluence at crack location.
4. Steady-state normal operating stresses to be used to calculate IASCC crack growth rates.
    a. Vertical stresses based on normal mechanical loads are required for horizontal flaws.
    b. Horizontal stresses based on normal mechanical loads are required for Vertical flaws.
    c. Stresses which have an insignificant net-through-wall value (average stress is near zero), such as weld residual stresses and thermal stresses due to local through-wall temperature gradients are considered to have minimal impact on the effective crack growth rates in through-wall flaws.
    d. Secondary weld residual and thermal stresses need to be considered in determination of circumferential and through-wall crack growth rates in partial through-wall flaws, whose dimensions would have to be determined with supplemental UT examinations.
5. Limiting externally applied transient stresses to be used to calculate allowable flaw lengths. For example, stresses arising from pressure, mechanical and thermal loads would be included in this calculation.



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**CE-ID: 3.1** **Core Shroud Assembly (Welded)**

**Remaining Axial Welds**

a. More detailed load-deformation histories may be required for elastic-plastic or limit load calculations, if these calculations are necessary.

Analysis: All analyses require an assumption of the IASCC crack growth expected over the upcoming period of service. The methodology for analysis of a one-sided visual examination is based on an assumption of a through-wall flaw with weld residual and thermal stresses relieved. An appropriate crack growth rate model will be applied, which may be either K dependent or K independent. In order to apply the acceptance criteria to a period beyond one refueling cycle (not to exceed a 10-year re-inspection interval), follow-up action is required to verify the assumptions used in the predicted crack-growth rate. A re-inspection of the indication at a future specified outage, for example, would provide data that could be used to satisfy this verification requirement.

Failure of the weld is assumed to occur when unstable crack growth is initiated from the analyzed flaw. Three options are outlined for determining the limiting allowable flaw length, based on neutron dose. Analysis methods are suggested for both pre-inspection and generic analysis (Suggested Pre-Inspection Analysis) and for flaws observed in-service (Suggested Flaw Specific Analysis), where more detailed characteristics of the flaw and its location are known. In all cases, a more detailed evaluation may be completed using a semi-elliptic surface flaw, but such an evaluation would require more detailed inspection by UT.



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## CE-ID: 3.1 Core Shroud Assembly (Welded)

### Remaining Axial Welds

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

Note:

1. Alternate fracture toughness values may be applied and justification for the alternate basis shall be submitted to the NRC for information within one year.
2. MRP-211 [66] provides the basis for the fracture toughness value.

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

Option 1: LEFM Analysis

a. Establish initial crack length (and depth if determined by supplemental examinations) based on inspection results.
b. The final crack dimensions are calculated by adding 10 years of crack growth under normal loading conditions.
c. Ensure that the $\Delta K$ resulting from flow-induced vibration (FIV) is below the threshold for fatigue crack growth.
d. For primary loading at normal and upset loading conditions, the final stress intensity factor (K) for the flaw size from (b) must be lower than the fracture toughness for the material by a factor of at least 2.77. For the evaluation of partial through wall flaws, the stress intensity factor due to secondary loading does not require an additional safety factor and may be directly added to the adjusted stress intensity factor for primary loading (i.e., $2.77 \cdot K_P + K_S < K_{IC}$).



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**CE-ID: 3.1** **Core Shroud Assembly (Welded)**

**Remaining Axial Welds**

e. For the governing primary loading at emergency or faulted loading conditions, the final stress intensity factor (K) for the flaw size from (b) must be lower than the fracture toughness for the material by a factor of at least 1.39. For the evaluation of partial through wall flaws, the stress intensity factor due to secondary loading does not require an additional safety factor and may be directly added to the adjusted stress intensity factor for primary loading (i.e., $1.39 \cdot K_P + K_S < K_{IC}$).

Option 2: EPFM Analysis (for neutron dose levels > 0.5 dpa but < 5 dpa)

a. Establish initial crack length (and depth if determined by supplemental examinations) based on inspection results.
b. The final crack dimensions are calculated by adding 10 years of crack growth under normal loading conditions.
c. Ensure that the ΔK resulting from flow-induced vibration (FIV) is below the threshold for fatigue crack growth.
d. Develop the (J/T) material curve from the material J-R curve.
e. The $J_{applied}$ curve must include a safety factor of 2.77 for primary loading at normal and upset conditions, and a factor of 1.39 for emergency and faulted conditions. For the evaluation of partial through wall flaws, the J curve due to secondary loading does not require an additional safety factor and may be directly added to the adjusted J curve for primary loading (i.e., $[2.77 \text{ or } 1.39] \cdot J_p + J_s < J_{applied}$).
f. The intersection of the material and applied (J/T) curves indicates the instability point. The flaw size at instability is determined from the $J_{applied}$ versus flaw size curve.
g. The flaw size at instability must be larger than the flaw size from (b), for the flaw to be acceptable.

The intersection of the material and applied (J/T) curves indicates the instability point. The load at instability is determined from the $J_{applied}$ versus load curve.

Option 3: Limit Load Analysis (For neutron fluence < $3 \times 10^{20}$ n/cm$^2$ at E>1 MeV or dose less than approximately 0.5 dpa). This option is unlikely due to the anticipated fluence levels.

a. Establish initial crack length (and depth if determined by supplemental examinations) based on inspection results.
b. The final crack dimensions are calculated by adding 10 years of crack growth under normal loading conditions.
c. Ensure that the ΔK resulting from flow-induced vibration (FIV) is below the threshold for fatigue crack growth.
d. Determine the bending moment (M) that can be tolerated as a function of the postulated flaw length.



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**CE-ID: 3.1** **Core Shroud Assembly (Welded)**

**Remaining Axial Welds**

e. The applied moment, increased by a factor of 1.39 (for primary loading at emergency and faulted conditions) or 2.77 (for primary loading at normal and upset conditions) must be less than the limit moment (from step d) for the flaw length determined in (b), for the flaw to be acceptable. For the evaluation of partial through wall flaws, the moment due to secondary loading does not require an additional safety factor and may be directly added to the adjusted moment for primary loading (i.e., $[2.77 \text{ or } 1.39] \cdot M_p + M_s < M_{applied}$).

| | |
|---|---|
| Acceptance Criteria: | The welds continue to perform its functional requirements with the projected flaw length at the end of the inspection interval. |
| Approach: | Limit load, EPFM, or LEFM analysis |
| Additional Action(s): | Note that the licensee shall submit the plant-specific fracture mechanics analysis justifying the detected flaws within one year after any inspection that detects relevant conditions as defined in Table 5-2 of MRP-227. |



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| CE-ID: 3.2 | Core Shroud Assembly (Welded)<br>Ribs and Rings | | |
|---|---|---|---|
| Category: | Expansion | Applicability: | Plant designs with core shrouds assembled with full-height shroud plates |
| Degradation Effect: | See MRP-227 Table 4-5 | | |
| Primary Link: | See MRP-227 Table 4-5 | | |
| Function: | Maintain dimensional stability of core shroud | | |

**Inspection**

| | |
|---|---|
| Method: | Inaccessible. See MRP-227 Table 4-5. |
| Coverage: | See MRP-227 Table 4-5. |
| Observable Effect: | As predicted by functionality analysis |

**Inputs and Assumptions**

It is assumed that the ribs and rings remain inaccessible (i.e., no advances occur to the inspection methods that would enable visual inspection of the ribs and rings).

**Failure**

| | |
|---|---|
| Failure Mechanism: | Cracking (IASCC) |
| Failure Effect: | 1. Core damage caused by event – require maintenance of coolable geometry.<br>2. Damage to peripheral fuel assemblies.<br>3. Through-wall crack provides leak path through shroud.<br>4. Potential loose parts. |
| Failure Criteria: | Maintain structural integrity of the core shroud |

**Methodology**

| | |
|---|---|
| Goal: | Demonstrate that structural integrity of the core shroud is maintained with degradation to the ribs and rings. Determine if observable effects are expected due to degradation of the ribs and rings. |
| Data Requirements: | 1. Geometry, loading, and fast neutron fluence of baffle-former assembly<br>2. Inspection results at locations on the core shroud that can be visually examined |
| Analysis: | Perform a functionality analysis of the core shroud assembly considering degradation of the supporting ribs and rings. Evaluate structural integrity in the degraded condition, and predict observable effects. |
| Acceptance Criteria: | Not applicable to inaccessible component. |



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



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| CE-ID: 4 | Core Shroud Assembly (Bolted) | | |
|---|---|---|---|
| | **Assembly** | | |
| Category: | Primary | Applicability: | Bolted plan designs |
| Degradation Effect: | See MRP-227 Table 4-2. | | |
| Expansion Link: | See MRP-227 Table 4-2. | | |
| Function: | Provide guidance and protection for the reactor core.<br>Provide a passageway for the distribution of the reactor coolant flow to the reactor core.<br>Provide gamma and neutron shielding for the reactor vessel. | | |
| **Inspection** | | | |
| Method: | Visual examination. See MRP-227 Table 4-2. | | |
| Coverage: | See MRP-227 Table 4-2. | | |
| Observable Effect: | Degradation of general condition as described above. | | |
| **Failure** | | | |
| Failure Mechanism: | Void swelling | | |
| Failure Effect: | 1. Interference with fuel assemblies<br>2. Obstruction of coolant flow<br>3. Loose parts generation<br>4. Distortion/misalignment of core<br>5. Local temperature peaks<br>6. Degradation of control element insertability | | |
| Failure Criteria: | No relevant observations | | |
| **Methodology** | | | |
| Goal: | A plant-specific plan should be developed for evaluating and mitigating the potential relevant conditions. The evaluation should consider any previously reported observations. | | |
| Data Requirements: | Baseline data on previous visual examinations of core shroud.<br>Performance records for peripheral fuel assemblies.<br>Loose parts monitoring data. | | |
| Analysis: | To ensure that the generically based conclusions presented are applicable to a plant-specific configuration, individual plants will need to confirm the following:<br>1. A VT-3 inspection of the core shroud assembly (bolted) is included in plant-specific inspections.<br>2. No indications or relevant condition have been observed in inspections completed to date for the assembly as part of the existing plant operating experience. | | |



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**CE-ID: 4** **Core Shroud Assembly (Bolted)**

**Assembly**

3. Modifications on MRP-227 components, with specific focus on Primary, Expansion, and Existing components, made to reactor internals prior to May 2007 are fully aligned with the MRP-227 strategy for managing aging in reactor internals.
4. Modifications that have been completed that would have a potential to impact the reactor internals since May 2007 have been evaluated and found to have no impact on MRP strategy and conclusions for managing aging in reactor internals.

Prepare for examination by conducting a failure modes and effects analysis (FMEA) to identify full range of potential relevant observations prior to inspection. Failure mechanisms considered should include:

- Broken or missing locking devices,
- Protruding bolt heads
- Missing bolts or bolt heads.
- Distortion or displacement of shroud plates
- Gross cracking of shroud plates
- Gaps at plate joints
- Interaction with fuel assemblies

| | |
|---|---|
| Acceptance Criteria: | Determined by FMEA |
| Approach: | FMEA should address plant-specific practices and priorities. Some generic work possible to outline issues and options to be addressed in FMEA. |
| Additional Action(s): | Note that the licensee shall submit the plant-specific or generic FMEA analysis and confirmation of applicability of generic FMEA analysis within one year after any inspection that detects relevant conditions as defined in Table 5-2 of MRP-227 |



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| | | | |
|---|---|---|---|
| **CE-ID: 4a** | **Core Shroud Assembly (Welded)** | | |
| | **Assembly** | | |
| Category: | Primary | Applicability: | Plant designs with core shrouds assembled in two vertical sections |
| Degradation Effect: | See MRP-227 Table 4-2. | | |
| Expansion Link: | See MRP-227 Table 4-2. | | |
| Function: | Provide support, guidance, and protection for the reactor core.<br>Provide a passageway for the distribution of the reactor coolant flow to the reactor core.<br>Provide gamma and neutron shielding for the reactor vessel. | | |
| **Inspection** | | | |
| Method: | Visual examination. See MRP-227 Table 4-2. | | |
| Coverage: | See MRP-227 Table 4-2. | | |
| Observable Effect: | Seam between upper and lower sections should appear even and consistent with any historical records.<br>Evidence of gaping between plates at protruding corners should be considered a relevant condition. | | |
| **Failure** | | | |
| Failure Mechanism: | Void Swelling | | |
| Failure Effect: | 1. Potential leakage through shroud.<br>2. Significant distortion may interfere with peripheral fuel assemblies.<br>3. Condition is a precursor to high stresses and potential cracking at weld seams.<br>4. Degradation of control element insertability | | |
| Failure Criteria: | 1. Damage on corresponding peripheral fuel assemblies.<br>2. Visible gap between the upper and lower section of the core shroud assembly | | |
| **Methodology** | | | |
| Goal: | A plant-specific plan should be developed for evaluating and mitigating the potential relevant conditions. The evaluation should consider any previously reported observations. | | |
| Data Requirements: | 1. Gap size<br>2. Swelling deformation model of shroud<br>3. Shroud fluence distribution<br>4. Shroud temperature distribution | | |



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| | |
|---|---|
| **CE-ID: 4a** | **Core Shroud Assembly (Welded)** |
| | **Assembly** |
| Analysis: | To ensure that the generically based conclusions presented are applicable to a plant-specific configuration, individual plants will need to confirm the following:<br>1. A VT-3 inspection of the Core Shroud Assembly (Welded) is included in plant-specific inspections.<br>2. No indications or relevant condition have been observed in inspections completed to date for the assembly as part of the existing plant operating experience.<br>3. Modifications on MRP-227 components, with specific focus on Primary, Expansion, and Existing components, made to reactor internals prior to May 2007 are fully aligned with the MRP-227 strategy for managing aging in reactor internals.<br>4. Modifications that have been completed that would have a potential to impact the reactor internals since May 2007 have been evaluated and found to have no impact on MRP strategy and conclusions for managing aging in reactor internals.<br>Prepare for examination by conducting a failure modes and effects analysis (FMEA) to identify full range of potential relevant observations prior to inspection. Failure mechanisms considered should include:<br>• Distortion or displacement of shroud plates<br>• Gross cracking of shroud plates<br>• Gaps at plate joints<br>• Interaction with fuel assemblies<br>Quantitative evaluation of swelling would require a time-dependent structural model that incorporates the effects of void swelling. Temperature gradients caused by gamma heating must be accurately estimated to provide reliable swelling estimates. This detailed evaluation is only required if repeated observations indicate that the gap is actively growing. |
| Acceptance Criteria: | Determined by FMEA |
| Approach: | Generic efforts to support inspection.<br>• Extension of MRP model to look at relationship between swelling and deformation at seam.<br>• Guideline for issues to be addressed in plant-specific FMEA. |
| Additional Action(s): | Note that the licensee shall submit the plant-specific or generic FMEA analysis and confirmation of applicability of generic FMEA analysis within one year after any inspection that detects relevant conditions as defined in Table 5-2 of MRP-227. |



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| **CE-ID: 5** | **Core Support Barrel Assembly**<br>**Upper Flange Weld** | | |
|---|---|---|---|
| Category: | Primary | Applicability: | All plants |
| Degradation Effect: | See MRP-227 Table 4-2 | | |
| Expansion Link: | See MRP-227 Table 4-2. | | |
| Function: | Primary core support structure | | |

**Inspection**

| | |
|---|---|
| Method: | Visual examination. See MRP-227 Table 4-2. |
| Coverage: | See MRP-227 Table 4-2. |
| Observable Effect: | The specific relevant condition is a detectable crack-like surface indication. |

**Inputs and Assumptions**

There are several inputs and assumptions that are critical to the development of acceptance criteria for the RV internals upper flange weld locations. These items are stated below:

- The inspections identified in MRP-227 are intended to provide a sampling of potential locations of degradation. Under this approach, inspection of one side (surface) of the weld is assumed to provide an adequate sampling for monitoring of SCC.
- The change in resistance to fracture of the RV upper flange weld can be correlated to the accumulated fluence at each weld location. Welds that are subject to low fluence are considered to have a high degree of resistance to fracture. Correspondingly, those welds subject to high fluence have lower resistance to fracture.
- It is assumed that the primary mode of crack growth to be expected in this region is stress corrosion cracking.
- The prediction of crack growth is based on the stress intensity factor, K, calculated using linear elastic fracture mechanics. The rate of crack growth is dependent on the amount of neutron fluence that the weld is expected to accumulate over the licensed operating lifetime. Since there has been very limited experience of SCC initiated cracks in operating PWRs to date, growth rates developed for the prediction of SCC in boiling water reactors (BWRs) are assumed to be appropriate for prediction of crack growth due to SCC in PWR reactor internals.
    - For weld locations subjected to fluence levels less than or equal to $5x10^{20}$ n/cm$^2$ (E>1MeV)(0.75 dpa), the maximum of the crack growth rate in 2017 ASME Section XI Appendix C-8520 **or** the crack growth equation in ASME Code Case N-889 [52] will be used. The conditions for the use of Code Case N-889 identified in Regulatory Guide RG 1.147, Rev. 20 [67] must be met.



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| | |
|---|---|
| **CE-ID: 5** | **Core Support Barrel Assembly**<br>**Upper Flange Weld** |
| | ○ For fluence levels above $5x10^{20}$ n/cm$^2$ (E>1MeV)(0.75 dpa), up to and including $1.4x10^{22}$ n/cm$^2$ (E>1 MeV) (20 dpa), the crack growth equation in ASME Code Case N-889 [52] will be used. The conditions for the use of Code Case N-889 identified in Regulatory Guide RG 1.147, Rev. 20 [67] must be met for the development of acceptance criteria.<br>• Depending on the magnitude of cyclic loads, such as thermal transients, additional crack growth as a result of these loads may need to be considered<br>Acceptance criteria can be developed for the entire operating license of a given plant by using predicted end-of-license fluence values. However, if there are changes to these fluence projections, such as in the event of a power uprate or change in core loading pattern, it would be necessary to confirm that the inputs selected based on fluence, such as SCC growth rate and fracture toughness, remain applicable until the end of the operating license. |
| **Failure** | |
| Failure Mechanism: | Cracking (SCC) |
| Failure Effect: | Potential loss–of-core support. |
| Failure Criteria: | Actively growing through-wall flaws may require mitigation.<br>An existing flaw is unacceptable if the flaw length projected at the next inspection cycle results in a potential loss-of-core support. |
| **Methodology** | |
| Goal: | Demonstrate that the cracking mechanism will not result in growth beyond the allowable crack length over the planned inspection interval. |
| Data Requirements: | 1. Surface crack length, as determined by visual inspection:<br>a. The 2013 edition of the ASME Code Section XI: IWA-3330(a) and Figure IWA-3330-1 provide requirements for the minimum allowable separation distance that will allow multiple cracks to be treated individually. This approach is part of the guidance in this report. The requirement in Section XI is that the ligament between adjacent cracks must be greater than half of the thickness of the material. If this criterion is not met, the individual crack lengths and the length of the ligament between the cracks must be summed. This total length is then compared to the allowable length. Note that the 2007 and 2010 editions of ASME Code Section XI have the equivalent content to the 2013 edition and are also applicable.<br>b. The flaw proximity rules in BWRVIP-158 [62] may be optionally applied in lieu of those defined in ASME Section XI: IWA-3330(a) and Figure IWA-3330-1 when appropriate. |



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**CE-ID: 5** **Core Support Barrel Assembly**

**Upper Flange Weld**

2. Flaw Depth
    a. For one-sided visual inspections, the flaw is assumed to be through-wall.
    b. Supplemental examinations may be used to determine depths of specific flaws as the basis for a flaw-specific evaluation, if needed.
3. Confirmation that fast neutron fluence at upper flange weld is below $3x10^{20}$ n/cm$^2$ (E> 1 MeV).
4. Steady-state normal operating stresses to be used to calculate SCC crack growth rates.
    a. Axial stresses based on normal mechanical loads are required for circumferential flaws.
    b. Stresses which have an insignificant net-through-wall value (average stress is near zero), such as weld residual stresses and thermal stresses due to local through-wall temperature gradients are considered to have minimal impact on the effective crack growth rates in through-wall flaws.
    c. Secondary weld residual and thermal stresses need to be considered in determination of circumferential and through-wall crack growth rates in partial through-wall flaws, whose dimensions would have to be determined with supplemental UT examinations.
5. Limiting externally applied transient stresses to be used to calculate allowable flaw lengths. Stresses arising from pressure, mechanical and thermal loads would be included in this calculation.
    a. More detailed load-deformation histories may be required for elastic-plastic or limit load calculations, if these calculations are necessary.

Analysis:

All analyses require an assumption of the SCC crack growth expected over the upcoming period of service. The methodology for analysis of a one-sided visual examination is based on an assumption of a through-wall flaw with weld residual and thermal stresses relieved. An appropriate crack growth rate model will be applied, which may be either K dependent or K independent. Fatigue crack growth has been assumed to be negligible. In order to apply the acceptance criteria to a period beyond one refueling cycle (not to exceed a 10-year re-inspection interval), additional action is required. The flaw evaluation will address the verification process of the predicted crack growth rate. Depending on the flaw size and knowledge of the plant conditions, a re-inspection at the next refueling outage may be required to provide data needed to operate beyond one cycle. The verification plan shall be included in the evaluation which is submitted to the regulator for their information. For detailed crack growth models used, see “Inputs and Assumptions” above.

Failure of the upper flange weld is assumed to occur when unstable circumferential crack growth is initiated from the analyzed flaw. Two options are outlined for determining the limiting allowable flaw length, based on neutron



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**CE-ID: 5** **Core Support Barrel Assembly**

**Upper Flange Weld**

dose. Analysis methods are suggested for both pre-inspection or generic analysis (Suggested Pre-Inspection Analysis) and for flaws observed in-service (Suggested Flaw Specific Analysis), where more detailed characteristics of the flaw and its location are known. In all cases, a more detailed evaluation may be completed using a semi-elliptic surface flaw, but such an evaluation would require more detailed inspection by UT.

| Fluence Range ($n/cm^2$ E>1MeV) | Dose (dpa) | Suggested Pre-Inspection Analysis | Suggested Flaw Specific Analysis |
|---|---|---|---|
| $\leq 3 \times 10^{20}$ | $\leq 0.5$ | LEFM using 150 ksi$\sqrt{in}$ as fracture toughness(1) or Limit Load | Limit Load |

Note:

1. Alternate fracture toughness values may be applied and justification for the alternate basis shall be submitted to the NRC for information within one year.

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

Option 1: LEFM Analysis

a. Establish initial crack length (and depth if determined by supplemental examinations) based on inspection results.
b. The final crack dimensions are calculated by adding 10 years of crack growth under normal loading conditions.
c. Ensure that the $\Delta K$ resulting from flow-induced vibration (FIV) is below the threshold for fatigue crack growth.
d. For primary loading at normal and upset loading conditions, the final stress intensity factor (K) for the flaw size from (b) must be lower than the fracture toughness for the material by a factor of at least 2.77. For the evaluation of partial through wall flaws, the stress intensity factor due to secondary loading does not require an additional safety factor and may be directly added to the adjusted stress intensity factor for primary loading (i.e., $2.77 \cdot K_P + K_S < K_{IC}$).
e. For the governing primary loading at emergency or faulted loading conditions, the final stress intensity factor (K) for the flaw size from (b) must be lower than the fracture toughness for the material by a factor of at least 1.39. For the evaluation of partial through wall flaws, the stress intensity factor due to secondary loading does not require an additional safety factor and may be directly added to the adjusted stress intensity factor for primary loading (i.e., $1.39 \cdot K_P + K_S < K_{IC}$).



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**CE-ID: 5** **Core Support Barrel Assembly**

**Upper Flange Weld**

Option 2: Limit Load Analysis (For neutron fluence < $3x10^{20}$ n/cm$^2$ at E>1 MeV or less than approximately 0.5 dpa)

a. Establish initial crack length (and depth if determined by supplemental examinations) based on inspection results.

b. The final crack dimensions are calculated by adding 10 years of crack growth under normal loading conditions.

c. Ensure that the ΔK resulting from flow-induced vibration (FIV) is below the threshold for fatigue crack growth.

d. Determine the bending moment (M) that can be tolerated as a function of the postulated flaw length.

e. The applied moment, increased by a factor of 1.39 (for primary loading at emergency and faulted conditions) or 2.77 (for primary loading at normal and upset conditions) must be less than the limit moment (from step d) for the flaw length determined in (b), for the flaw to be acceptable. For the evaluation of partial through wall flaws, the moment due to secondary loading does not require an additional safety factor and may be directly added to the adjusted moment for primary loading (i.e., [2.77 or 1.39]·$M_p$ + $M_s$ < $M_{applied}$).

Acceptance Criteria: The upper flange weld continues to perform its functional requirements with the projected flaw length at the end of the inspection interval.

Approach: Limit load or LEFM analysis

Additional Action(s):

1) In order to apply the TR methodology for cracks detected via one-sided visual examinations, if supplementary examinations to determine crack depth were not performed, a follow up examination must be performed no later than the next RFO to confirm the assumed crack growth rates are conservative, unless a technical justification for a longer verification interval is submitted.

2) Note that the licensee shall submit the plant-specific crack growth and fracture mechanics analysis justifying the detected flaws, including a justification of the allowable operating period before re-inspection, within one year after any inspection that triggers the expansion criteria as defined in Table 5-2 of MRP-227.



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| CE-ID: 5.1 | Core Support Barrel Assembly<br>Lower Girth Weld | | |
|---|---|---|---|
| Category: | Expansion | Applicability: | All plants |
| Degradation Effect: | See MRP-227 Table 4-5 | | |
| Primary Link: | See MRP-227 Table 4-5 | | |
| Function: | Primary core support structure | | |

**Inspection**

| | |
|---|---|
| Method: | Visual examination. See MRP-227 Table 4-5. |
| Coverage: | See MRP-227 Table 4-5. |
| Observable Effect: | The specific relevant condition is a detectable crack-like surface indication. |

**Inputs and Assumptions**

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

- The inspections identified in MRP-227 are intended to provide a sampling of potential locations of degradation. Under this approach, inspection of one side (surface) of the weld is assumed to provide an adequate sampling for monitoring of SCC.
- It is assumed that these welds experience low fluence. The change in resistance to fracture of the RV lower girth weld can be correlated to the accumulated fluence at each weld location. Welds that are subject to low fluence are considered to have a high degree of resistance to fracture. Correspondingly, those welds subject to high fluence have lower resistance to fracture.
- It is assumed that the primary mode of crack growth to be expected in this region is stress corrosion cracking.
- The prediction of crack growth is based on the stress intensity factor, K, calculated using linear elastic fracture mechanics. The rate of crack growth is dependent on the amount of neutron fluence that the weld is expected to accumulate over the licensed operating lifetime. Since there has been very limited experience of SCC initiated cracks in operating PWRs to date, growth rates developed for the prediction of SCC in boiling water reactors (BWRs) are assumed to be appropriate for prediction of crack growth due to SCC in PWR reactor internals.
    - For weld locations subjected to fluence levels less than or equal to $5x10^{20}$ n/cm$^2$ (E>1MeV)(0.75 dpa), the maximum of the crack growth rate in 2017 ASME Section XI Appendix C-8520 **or** the crack growth equation in ASME Code Case N-889 [52] will be used. The conditions for the use of Code Case N-889 identified in Regulatory Guide RG 1.147, Rev. 20 [67] must be met.
    - For fluence levels above $5x10^{20}$ n/cm$^2$ (E>1MeV)(0.75 dpa) up to and including $1.4x10^{22}$ n/cm$^2$ (E>1 MeV) (20 dpa), the crack growth equation in ASME Code Case N-889 [52] will be used.



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| | |
|---|---|
| **CE-ID: 5.1** | **Core Support Barrel Assembly**<br>**Lower Girth Weld** |
| | The conditions for the use of Code Case N-889 identified in Regulatory Guide RG 1.147, Rev. 20 [67] must be met for the development of acceptance criteria.<br>• Depending on the magnitude of cyclic loads, such as thermal transients, additional crack growth as a result of these loads may need to be considered.<br>Acceptance criteria can be developed for the entire operating license of a given plant by using predicted end-of-license fluence values. However, if there are changes to these fluence projections, such as in the event of a power uprate or change in core loading pattern, it would be necessary to confirm that the inputs selected based on fluence, such as SCC growth rate and fracture toughness, remain applicable until the end of the operating license. |
| **Failure** | |
| Failure Mechanism: | Cracking (SCC, Fatigue) |
| Failure Effect: | Potential loss-of-core support. |
| Failure Criteria: | Actively growing through-wall flaws may require mitigation.<br>An existing flaw is unacceptable if the flaw length projected at the next inspection cycle allows a potential loss-of-core support. |
| **Methodology** | |
| Goal: | Demonstrate that the cracking mechanism will not result in growth beyond the allowable crack length over the planned inspection interval. |
| Data Requirements: | 1. Surface crack length as determined by visual inspection<br>a. The 2013 edition of the ASME Code Section XI: IWA-3330(a) and Figure IWA-3330-1 provide requirements for the minimum allowable separation distance that will serve as the basis for guidance in this report. The requirement in Section XI is that the ligament between adjacent cracks must be greater than half of the thickness of the material. If this criterion is not met, the individual crack lengths and the length of the ligament between the cracks must be summed. This total length is then compared to the allowable length. Note that the 2007 and 2010 editions of ASME Code Section XI have the equivalent content to the 2013 edition and are also applicable.<br>b. The flaw proximity rules in BWRVIP-158 [62] may be optionally applied in lieu of those defined in ASME Section XI: IWA-3330(a) and Figure IWA-3330-1 when appropriate.<br>2. Flaw Depth<br>a. For one-sided visual inspections, the flaw is assumed to be through-wall.<br>b. Supplemental examinations may be used to determine flaw depth, for a flaw-specific criterion, if needed. |



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**CE-ID: 5.1** **Core Support Barrel Assembly**

**Lower Girth Weld**

3. Estimate of fast neutron fluence at crack location.
4. Steady-state normal operating stresses to be used to calculate SCC crack growth rates.
    a. Axial stresses based on normal mechanical loads are required for circumferential flaws.
    b. Stresses which have an insignificant net-through-wall value (average stress is near zero), such as weld residual stresses and thermal stresses due to local through-wall temperature gradients are considered to have minimal impact on the effective crack growth rates in through-wall flaws.
    c. Secondary weld residual and thermal stresses need to be considered in determination of circumferential and through-wall crack growth rates in partial through-wall flaws, whose dimensions would have to be determined with supplemental UT examinations.
5. Limiting externally applied transient stresses to be used to calculate allowable flaw lengths. Stresses arising from pressure, mechanical and thermal loads would be included in this calculation.
    a. More detailed load-deformation histories may be required for elastic-plastic or limit load calculations, if these calculations are necessary.

Analysis:

All analyses require an assumption of the SCC crack growth expected over the upcoming period of service. The methodology for analysis of a one-sided visual examination is based on an assumption of a through-wall flaw with weld residual and thermal stresses relieved. An appropriate crack growth rate model will be applied, which may be either K dependent or K independent. Fatigue crack growth has been assumed to be negligible. In order to apply the acceptance criteria to a period beyond one refueling cycle (not to exceed a 10-year re-inspection interval), additional action is required. The flaw evaluation will address the verification process of the predicted crack growth rate. Depending on the flaw size and knowledge of the plant conditions, a re-inspection at the next refueling outage may be required to provide data needed to operate beyond one cycle. The verification plan shall be included in the evaluation which is submitted to the regulator for their information. For detailed crack growth models used, see “Inputs and Assumptions” above.

Failure of the lower girth weld is assumed to occur when unstable circumferential crack growth is initiated from the analyzed flaw. Two options are outlined for determining the limiting allowable flaw length, based on neutron dose. Analysis methods are suggested for both pre-inspection and generic analysis (Suggested Pre-Inspection Analysis) and for flaws observed in-service (Suggested Flaw Specific Analysis), where more detailed characteristics of the flaw and its location are known. In all cases, a more detailed evaluation may be completed using a semi-elliptic surface flaw, but such an evaluation would require more detailed inspection by UT.



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**CE-ID: 5.1** **Core Support Barrel Assembly**

**Lower Girth Weld**

| Fluence Range ($n/cm^2$ E>1MeV) | Dose (dpa) | Suggested Pre-Inspection Analysis | Suggested Flaw Specific Analysis |
|---|---|---|---|
| $\leq 3 \times 10^{20}$ | $\leq 0.5$ | LEFM using 150 ksi$\sqrt{in}$ as fracture toughness(1) or Limit Load | Limit Load |

Note:

1. Alternate fracture toughness values may be applied and justification for the alternate basis shall be submitted to the NRC for information within one year.

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

Option 1: LEFM Analysis

a. Establish initial crack length (and depth if determined by supplemental examinations) based on inspection results.
b. The final crack dimensions are calculated by adding 10 years of crack growth under normal loading conditions.
c. Ensure that the $\Delta K$ resulting from flow-induced vibration (FIV) is below the threshold for fatigue crack growth.
d. For primary loading at normal and upset loading conditions, the final stress intensity factor (K) for the flaw size from (b) must be lower than the fracture toughness for the material by a factor of at least 2.77. For the evaluation of partial through wall flaws, the stress intensity factor due to secondary loading does not require an additional safety factor and may be directly added to the adjusted stress intensity factor for primary loading (i.e., $2.77 \cdot K_P + K_S < K_{IC}$).
e. For the governing primary loading at emergency or faulted loading conditions, the final stress intensity factor (K) for the flaw size from (b) must be lower than the fracture toughness for the material by a factor of at least 1.39. For the evaluation of partial through wall flaws, the stress intensity factor due to secondary loading does not require an additional safety factor and may be directly added to the adjusted stress intensity factor for primary loading (i.e., $1.39 \cdot K_P + K_S < K_{IC}$).

Option 2: Limit Load Analysis (For neutron fluence < $3 \times 10^{20}$ $n/cm^2$ at E>1 MeV or dose less than approximately 0.5 dpa)



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**CE-ID: 5.1** **Core Support Barrel Assembly**

**Lower Girth Weld**

a. Establish initial crack length (and depth if determined by supplemental examinations) based on inspection results.

b. The final crack dimensions are calculated by adding 10 years of crack growth under normal loading conditions.

c. Ensure that the $\Delta K$ resulting from flow-induced vibration (FIV) is below the threshold for fatigue crack growth.

d. Determine the bending moment (M) that can be tolerated as a function of the postulated flaw length.

e. The applied moment, increased by a factor of 1.39 (for primary loading at emergency and faulted conditions) or 2.77 (for primary loading at normal and upset conditions) must be less than the limit moment (from step d) for the flaw length determined in (b), for the flaw to be acceptable. For the evaluation of partial through wall flaws, the moment due to secondary loading does not require an additional safety factor and may be directly added to the adjusted moment for primary loading (i.e., $[2.77 \text{ or } 1.39] \cdot M_p + M_s < M_{applied}$).

Acceptance Criteria: The lower girth weld continues to perform its functional requirements with the projected flaw length at the end of the inspection interval.

Approach: Limit load or LEFM analysis

Additional Action(s):

1) In order to apply the TR methodology for cracks detected via one-sided visual examinations, if supplementary examinations to determine crack depth were not performed, a follow up examination must be performed no later than the next RFO to confirm the assumed crack growth rates are conservative, unless a technical justification for a longer verification interval is submitted.

2) Note that the licensee shall submit the plant-specific crack growth and fracture mechanics analysis justifying the detected flaws, including a justification of the allowable operating period before re-inspection, within one year after any inspection that detects relevant conditions as defined in Table 5-2 of MRP-227.



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| CE-ID: 5.2 | Core Support Barrel Assembly<br>Upper Girth Weld | | |
|---|---|---|---|
| Category: | Expansion | Applicability: | All plants |
| Degradation Effect: | See MRP-227 Table 4-5 | | |
| Primary Link: | See MRP-227 Table 4-5 | | |
| Function: | Primary core support structure | | |

**Inspection**

| | |
|---|---|
| Method: | Visual examination. See MRP-227 Table 4-5. |
| Coverage: | See MRP-227 Table 4-5. |
| Observable Effect: | The specific relevant condition is a detectable crack-like surface indication. |

**Inputs and Assumptions**

There are several inputs and assumptions that are critical to the development of acceptance criteria for the RV internals core support barrel upper girth weld. These items are stated below:

- The inspections identified in MRP-227 are intended to provide a sampling of potential locations of degradation. Under this approach, inspection of one side (surface) of the weld is assumed to provide an adequate sampling for monitoring of SCC.
- The change in resistance to fracture of the core support barrel upper girth weld can be correlated to the accumulated fluence at each weld location. Welds that are subject to low fluence are considered to have a high degree of resistance to fracture. Correspondingly, those welds subject to high fluence have lower resistance to fracture.
- It is assumed that the primary mode of crack growth to be expected in this region is stress corrosion cracking.
- The prediction of crack growth is based on the stress intensity factor, K, calculated using linear elastic fracture mechanics. The rate of crack growth is dependent on the amount of neutron fluence that the weld is expected to accumulate over the licensed operating lifetime. Since there has been very limited experience of SCC initiated cracks in in this or comparable components or welds in operating PWRs to date, growth rate models must be obtained from other sources. One source is the models developed for predicting SCC crack growth rate (CGR) in BWR components. The other source is the models developed based on laboratory data that may include test results for both PWR and BWR representative conditions. These models were considered to be appropriate for prediction of crack growth due to SCC in PWR reactor internals.
    - For weld locations subjected to fluence levels less than or equal to $5x10^{20}$ n/cm$^2$ (E>1MeV)(0.75 dpa), the maximum of the crack growth rate in 2017 ASME Section XI Appendix C-8520 **or** the crack growth equation in ASME Code Case N-889 [52] will be used. The conditions for the use of Code Case N-889 identified in Regulatory Guide RG 1.147, Rev. 20 [67] must be met.



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**CE-ID: 5.2** **Core Support Barrel Assembly**

**Upper Girth Weld**

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

- Depending on the magnitude of cyclic loads, such as thermal transients, additional crack growth as a result of these loads may need to be considered.
- Acceptance criteria for indications detected using a visual exam can be developed based on an assumed through-wall flaw with a length that is uniform through the wall thickness. When the critical crack length for a through wall flaw is less than that for a part through wall flaw, this assumption is reasonable and conservative, since no information is available on the flaw depth from such a visual examination.
- When developing flaw acceptance criteria based on a through-wall flaw, the crack growth due to SCC can be predicted using only the externally applied stresses. This simplification can be made because the average of the thermal and residual stresses across the crack face is zero. Therefore, these stresses are considered to have an insignificant contribution to the average crack growth across the through-wall crack front.
- Acceptance criteria can be developed for the entire operating license of a given plant by using predicted end-of-license fluence values. However, if there are changes to these fluence projections, such as in the event of a power uprate or change in core loading pattern, it would be necessary to confirm that the inputs selected based on fluence, such as SCC growth rate and fracture toughness, remain applicable until the end of the operating license.

| **Failure** | |
|---|---|
| Failure Mechanism: | Cracking (SCC) |
| Failure Effect: | Potential loss–of-core support. |
| Failure Criteria: | Actively growing through-wall flaws may require mitigation.<br>An existing flaw is unacceptable if the flaw length projected at the next inspection cycle results in a potential loss-of-core support. |

**Methodology**



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**CE-ID: 5.2** **Core Support Barrel Assembly**

**Upper Girth Weld**

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

Data Requirements:

1. Surface crack length as determined by visual inspection
    a. The 2013 edition of the ASME Code Section XI: IWA-3330(a) and Figure IWA-3330-1 provide requirements for the minimum allowable separation distance that will serve as the basis for guidance in this report. The requirement in Section XI is that the ligament between adjacent cracks must be greater than half of the thickness of the material. If this criterion is not met, the individual crack lengths and the length of the ligament between the cracks must be summed. This total length is then compared to the allowable length. Note that the 2007 and 2010 editions of ASME Code Section XI have the equivalent content to the 2013 edition and are also applicable.
    b. The flaw proximity rules in BWRVIP-158 [62] may be optionally applied in lieu of those defined in ASME Section XI: IWA-3330(a) and Figure IWA-3330-1 when appropriate.
2. Flaw Depth
    a. For one-sided visual inspections, the flaw is assumed to be through-wall.
    b. Supplemental examinations may be used to determine flaw depth, for a flaw-specific criterion, if needed.
3. Estimate of fast neutron fluence at crack location
4. Steady-state normal operating stresses to be used to calculate SCC crack growth rates.
    a. Axial stresses based on normal mechanical loads are required for circumferential flaws.
    b. Stresses which have an insignificant net-through-wall value (average stress is near zero), such as weld residual stresses and thermal stresses due to local through-wall temperature gradients are considered to have minimal impact on the effective crack growth rates in through-wall flaws.
    c. Secondary weld residual and thermal stresses need to be considered in determination of circumferential and through-wall crack growth rates in partial through-wall flaws, whose dimensions would have to be determined with supplemental UT examinations.
5. Limiting externally applied transient stresses to be used to calculate allowable flaw lengths. Stresses arising from pressure, mechanical and thermal loads would be included in this calculation.
    a. More detailed load-deformation histories may be required for elastic-plastic or limit load calculations, if these calculations are necessary.

Analysis: All analyses require an assumption of the SCC crack growth expected over the upcoming period of service. The methodology for analysis of a one-sided visual examination is based on an assumption of a through-wall flaw with weld residual



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**CE-ID: 5.2** **Core Support Barrel Assembly**

**Upper Girth Weld**

and thermal stresses relieved. An appropriate crack growth rate model will be applied, which may be either K dependent or K independent. Fatigue crack growth has been assumed to be negligible. In order to apply the acceptance criteria to a period beyond one refueling cycle (not to exceed a 10-year re-inspection interval), additional action is required. The flaw evaluation will address the verification process of the predicted crack growth rate. Depending on the flaw size and knowledge of the plant conditions, a re-inspection at the next refueling outage may be required to provide data needed to operate beyond one cycle. The verification plan shall be included in the evaluation which is submitted to the regulator for their information. For detailed crack growth models used, see "Inputs and Assumptions" above.

Failure of the upper girth weld is assumed to occur when unstable crack growth is initiated from the analyzed flaw. Three options are outlined for determining the limiting allowable flaw length. Analysis methods are suggested for both pre-inspection or generic analysis (Suggested Pre-Inspection Analysis) and for flaws observed in-service (Suggested Flaw Specific Analysis), where more detailed characteristics of the flaw and its location are known. In all cases, a more detailed evaluation may be completed using a semi-elliptic surface flaw, but such an evaluation would require more detailed inspection by UT.

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

Note:

1. Alternate fracture toughness values may be applied and justification for the alternate basis shall be submitted to the NRC for information within one year.
2. MRP-211 [66] provides the basis for the fracture toughness value.



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**CE-ID: 5.2** **Core Support Barrel Assembly**

**Upper Girth Weld**

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

Option 1: LEFM Analysis

a. Establish initial crack length (and depth if determined by supplemental examinations) based on inspection results.
b. The final crack dimensions are calculated by adding 10 years of crack growth under normal loading conditions.
c. Ensure that the ΔK resulting from flow-induced vibration (FIV) is below the threshold for fatigue crack growth.
d. For primary loading at normal and upset loading conditions, the final stress intensity factor (K) for the flaw size from (b) must be lower than the fracture toughness for the material by a factor of at least 2.77. For the evaluation of partial through wall flaws, the stress intensity factor due to secondary loading does not require an additional safety factor and may be directly added to the adjusted stress intensity factor for primary loading (i.e., $2.77 \cdot K_P + K_S < K_{IC}$).
e. For the governing primary loading at emergency or faulted loading conditions, the final stress intensity factor (K) for the flaw size from (b) must be lower than the fracture toughness for the material by a factor of at least 1.39. For the evaluation of partial through wall flaws, the stress intensity factor due to secondary loading does not require an additional safety factor and may be directly added to the adjusted stress intensity factor for primary loading (i.e., $1.39 \cdot K_P + K_S < K_{IC}$).

Option 2: EPFM Analysis (for neutron dose levels > 0.5 dpa but < 5 dpa)

a. Establish initial crack length (and depth if determined by supplemental examinations) based on inspection results.
b. The final crack dimensions are calculated by adding 10 years of crack growth under normal loading conditions.
c. Ensure that the ΔK resulting from flow-induced vibration (FIV) is below the threshold for fatigue crack growth.
d. Develop the (J/T) material curve from the material J-R curve.
e. The $J_{applied}$ curve must include a safety factor of 2.77 for primary loading at normal and upset conditions, and a factor of 1.39 for primary loading at emergency and faulted conditions. For the evaluation of partial through wall flaws, the J curve due to secondary loading does not require an additional safety factor and may be directly added to the adjusted J curve for primary loading (i.e., $[2.77 \text{ or } 1.39] \cdot J_p + J_s < J_{applied}$).
f. The intersection of the material and applied (J/T) curves indicates the instability point. The flaw size at instability is determined from the $J_{applied}$ versus flaw size curve.



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**CE-ID: 5.2** **Core Support Barrel Assembly**

**Upper Girth Weld**

g. The flaw size at instability must be larger than the flaw size from (b), for the flaw to be acceptable.

The intersection of the material and applied (J/T) curves indicates the instability point. The load at instability is determined from the $J_{applied}$ versus load curve.

Option 3: Limit Load Analysis (For neutron fluence < $3x10^{20}$ n/cm$^2$ at E>1 MeV or dose less than approximately 0.5 dpa)

a. Establish initial crack length (and depth if determined by supplemental examinations) based on inspection results.

b. The final crack dimensions are calculated by adding 10 years of crack growth under normal loading conditions.

c. Ensure that the ΔK resulting from flow-induced vibration (FIV) is below the threshold for fatigue crack growth.

d. Determine the bending moment (M) that can be tolerated as a function of the postulated flaw length.

e. The applied moment, increased by a factor of 1.39 (for primary loading at emergency and faulted conditions) or 2.77 (for primary loading at normal and upset conditions) must be less than the limit moment (from step d) for the flaw length determined in (b), for the flaw to be acceptable. For the evaluation of partial through wall flaws, the moment due to secondary loading does not require an additional safety factor and may be directly added to the adjusted moment for primary loading (i.e., $[2.77 \text{ or } 1.39]\cdot M_p + M_s < M_{applied}$).

Acceptance Criteria: The upper girth weld continues to perform its functional requirements with the projected flaw length at the end of the inspection interval.

Approach: Limit load, EPFM, or LEFM analysis

Additional Action(s):

1) In order to apply the TR methodology for cracks detected via one-sided visual examinations, if supplementary examinations to determine crack depth were not performed, a follow up examination must be performed no later than the next RFO to confirm the assumed crack growth rates are conservative, unless a technical justification for a longer verification interval is submitted.

2) Note that the licensee shall submit the plant-specific crack growth and fracture mechanics analysis justifying the detected flaws, including a justification of the allowable operating period before re-inspection, within one year after any inspection that detects relevant conditions as defined in Table 5-2 of MRP-227.



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| CE-ID: 5.3 | Core Support Barrel Assembly<br>Upper Axial Weld | | |
|---|---|---|---|
| Category: | Expansion | Applicability: | All plants |
| Degradation Effect: | See MRP-227 Table 4-5 | | |
| Primary Link: | See MRP-227 Table 4-5 | | |
| Function: | The upper axial welds contribute to the overall structural integrity of the core barrel. | | |

**Inspection**

| | |
|---|---|
| Method: | Visual examination. See MRP-227 Table 4-5. |
| Coverage: | See MRP-227 Table 4-5 |
| Observable Effect: | The specific relevant condition is a detectable crack-like surface indication. |

**Inputs and Assumptions**

There are several inputs and assumptions that are critical to the development of acceptance criteria for the RV internals core barrel assembly upper axial weld. These items are stated below:

- The inspections identified in MRP-227 are intended to provide a sampling of potential locations of degradation. Under this approach, inspection of one side (surface) of the weld is assumed to provide an adequate sampling for monitoring of SCC.
- The change in resistance to fracture of the RV upper axial weld can be correlated to the accumulated fluence at each weld location. Welds that are subject to low fluence are considered to have a high degree of resistance to fracture. Correspondingly, those welds subject to high fluence have lower resistance to fracture.
- It is assumed that the primary mode of crack growth to be expected in this region is stress corrosion cracking.
- The prediction of crack growth is based on the stress intensity factor, K, calculated using linear elastic fracture mechanics. The rate of crack growth is dependent on the amount of neutron fluence that the weld is expected to accumulate over the licensed operating lifetime. Since there has been very limited experience of IASCC initiated cracks in this or comparable components or welds in operating PWRs to date, growth rate models must be obtained from other sources. One source is the models developed for predicting IASCC crack growth rate (CGR) in BWR components. The other source is the models developed based on laboratory data that may include test results for both PWR and BWR representative conditions. These models were considered to be appropriate for the prediction of crack growth due to IASCC in PWR internals.
    - For weld locations subjected to fluence levels less than or equal to $5x10^{20}$ n/cm$^2$ (E>1MeV)(0.75 dpa), the maximum of the crack growth rate in 2017 ASME Section XI Appendix C-8520 **or** the crack growth equation in ASME Code Case N-889 [52] will be used. The conditions for the use of Code Case N-889 identified in Regulatory Guide RG 1.147, Rev. 20 [67] must be met.



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| | |
|---|---|
| **CE-ID: 5.3** | **Core Support Barrel Assembly**<br>**Upper Axial Weld** |

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

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

**Failure**

| | |
|---|---|
| Failure Mechanism: | Cracking (SCC) |
| Failure Effect: | Potential for stress redistribution within the core barrel, which could initiate or accelerate circumferential crack growth in the core barrel. |
| Failure Criteria: | Actively growing through-wall flaws may require mitigation.<br>An existing flaw is unacceptable if the flaw length projected at the next inspection cycle results in a potential loss-of-core support. |

**Methodology**

| | |
|---|---|
| Goal: | Demonstrate that the cracking mechanism will not result in growth beyond the allowable crack length over the planned inspection interval. |
| Data Requirements: | 1. Surface crack length as determined by visual inspection<br>a. The 2013 edition of the ASME Code Section XI: IWA-3330(a) and Figure IWA-3330-1 provide requirements for the minimum allowable separation distance that will serve as the basis for guidance in this report. The requirement in Section XI is that the ligament between adjacent cracks must be greater than half of the thickness of the material. If this criterion is not met, the individual crack lengths and the length of the ligament between the cracks must be summed. This total length is then compared to the allowable |



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**CE-ID: 5.3** **Core Support Barrel Assembly**

**Upper Axial Weld**

length. Note that the 2007 and 2010 editions of ASME Code Section XI have the equivalent content to the 2013 edition and are also applicable.

b. The flaw proximity rules in BWRVIP-158 [62] may be optionally applied in lieu of those defined in ASME Section XI: IWA-3330(a) and Figure IWA-3330-1 when appropriate.

2. Flaw Depth
   a. For one-sided visual inspections, the flaw is assumed to be through-wall.
   b. Supplemental examinations may be used to determine flaw depth, for a flaw-specific criterion, if needed.
3. Estimate of fast neutron fluence at crack location.
4. Steady-state normal operating stresses to be used to calculate SCC crack growth rates.
   a. Hoop stresses based on normal mechanical loads are required for axial flaws.
   b. Stresses which have an insignificant net-through-wall value (average stress is near zero), such as weld residual stresses and thermal stresses due to local through-wall temperature gradients are considered to have minimal impact on the effective crack growth rates in through-wall flaws.
   c. Secondary weld residual and thermal stresses need to be considered in determination of axial and through-wall crack growth rates in partial through-wall flaws, whose dimensions would have to be determined with supplemental UT examinations.
5. Limiting externally applied transient stresses to be used to calculate allowable flaw lengths. Stresses arising from pressure, mechanical and thermal loads would be included in this calculation.
   a. More detailed load-deformation histories may be required for elastic-plastic or limit load calculations, if these calculations are necessary.

Analysis: All analyses require an assumption of the SCC crack growth expected over the upcoming period of service. The methodology for analysis of a one-sided visual examination is based on an assumption of a through-wall flaw with weld residual and thermal stresses relieved. An appropriate crack growth rate model will be applied, which may be either K dependent or K independent. Fatigue crack growth has been assumed to be negligible. In order to apply the acceptance criteria to a period beyond one refueling cycle (not to exceed a 10-year re-inspection interval), additional action is required. The flaw evaluation will address the verification process of the predicted crack growth rate. Depending on the flaw size and knowledge of the plant conditions, a re-inspection at the next refueling outage may be required to provide data needed to operate beyond one cycle. The verification plan shall be included in the evaluation which is submitted to the regulator for their



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## CE-ID: 5.3 Core Support Barrel Assembly

### Upper Axial Weld

information. For detailed crack growth models used, see "Inputs and Assumptions" above.

Failure of the axial welds is assumed to occur when unstable axial crack growth is initiated from the analyzed flaw. Three options are outlined for determining the limiting allowable flaw length, based on neutron dose. Analysis methods are suggested for both pre-inspection or generic analysis (Suggested Pre-Inspection Analysis) and for flaws observed in-service (Suggested Flaw Specific Analysis), where more detailed characteristics of the flaw and its location are known. In all cases, a more detailed evaluation may be completed using a semi-elliptic surface flaw, but such an evaluation would require more detailed inspection by UT.

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

Note:

1. Alternate fracture toughness values may be applied and justification for the alternate basis shall be submitted to the NRC for information within one year.
2. MRP-211 [66] provides the basis for the fracture toughness value.

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



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**CE-ID: 5.3** **Core Support Barrel Assembly**

**Upper Axial Weld**

Option 1: LEFM Analysis

a. Establish initial crack length (and depth if determined by supplemental examinations) based on inspection results.

b. The final crack dimensions are calculated by adding 10 years of crack growth under normal loading conditions.

c. Ensure that the ΔK resulting from flow-induced vibration (FIV) is below the threshold for fatigue crack growth.

d. For primary loading at normal and upset loading conditions, the final stress intensity factor (K) for the flaw size from (b) must be lower than the fracture toughness for the material by a factor of at least 2.77. For the evaluation of partial through wall flaws, the stress intensity factor due to secondary loading does not require an additional safety factor and may be directly added to the adjusted stress intensity factor for primary loading (i.e., $2.77 \cdot K_P + K_S < K_{IC}$).

e. For the governing primary loading at emergency or faulted loading conditions, the final stress intensity factor (K) for the flaw size from (b) must be lower than the fracture toughness for the material by a factor of at least 1.39. For the evaluation of partial through wall flaws, the stress intensity factor due to secondary loading does not require an additional safety factor and may be directly added to the adjusted stress intensity factor for primary loading (i.e., $1.39 \cdot K_P + K_S < K_{IC}$).

Option 2: EPFM Analysis (for neutron dose levels > 0.5 dpa but < 5 dpa)

a. Establish initial crack length (and depth if determined by supplemental examinations) based on inspection results.

b. The final crack dimensions are calculated by adding 10 years of crack growth under normal loading conditions.

c. Ensure that the ΔK resulting from flow-induced vibration (FIV) is below the threshold for fatigue crack growth.

d. Develop the (J/T) material curve from the material J-R curve.

e. The $J_{applied}$ curve must include a safety factor of 2.77 for primary loading at normal and upset conditions, and a factor of 1.39 for primary loading at emergency and faulted conditions. For the evaluation of partial through wall flaws, the J curve due to secondary loading does not require an additional safety factor and may be directly added to the adjusted J curve for primary loading (i.e., $[2.77 \text{ or } 1.39] \cdot J_p + J_s < J_{applied}$).

f. The intersection of the material and applied (J/T) curves indicates the instability point. The flaw size at instability is determined from the $J_{applied}$ versus flaw size curve.

g. The flaw size at instability must be larger than the flaw size from (b), for the flaw to be acceptable.



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**CE-ID: 5.3** **Core Support Barrel Assembly**

**Upper Axial Weld**

The intersection of the material and applied (J/T) curves indicates the instability point. The load at instability is determined from the $J_{applied}$ versus load curve.

Option 3: Limit Load Analysis (For neutron fluence < $3x10^{20}$ n/cm$^2$ at E>1 MeV or dose less than approximately 0.5 dpa)

a. Establish initial crack length (and depth if determined by supplemental examinations) based on inspection results.

b. The final crack dimensions are calculated by adding 10 years of crack growth under normal loading conditions.

c. Ensure that the ΔK resulting from flow-induced vibration (FIV) is below the threshold for fatigue crack growth.

d. Determine the bending moment (M) that can be tolerated as a function of the postulated flaw length.

e. The applied moment, increased by a factor of 1.39 (for primary loading at emergency and faulted conditions) or 2.77 (for primary loading at normal and upset conditions) must be less than the limit moment (from step d) for the flaw length determined in (b), for the flaw to be acceptable. For the evaluation of partial through wall flaws, the moment due to secondary loading does not require an additional safety factor and may be directly added to the adjusted moment for primary loading (i.e., [2.77 or 1.39]·$M_p$ + $M_s < M_{applied}$).

Acceptance Criteria: The upper axial welds continue to perform their functional requirements with the projected flaw lengths at the end of the inspection interval.

Approach: Limit load, EPFM, or LEFM analysis

Additional Action(s):

1) In order to apply the TR methodology for cracks detected via one-sided visual examinations, if supplementary examinations to determine crack depth were not performed, a follow up examination must be performed no later than the next RFO to confirm the assumed crack growth rates are conservative, unless a technical justification for a longer verification interval is submitted.

2) Note that the licensee shall submit the plant-specific crack growth and fracture mechanics analysis justifying the detected flaws, including the justification of the allowable operating period before re-inspection, within one year after any inspection that detects relevant conditions as defined in Table 5-2 of MRP-227.



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| **CE-ID: 5.4** | **Lower Support Structure**<br>**Lower Core Support Beams** | | |
|---|---|---|---|
| Category: | Expansion | Applicability: | All plants except those with core shrouds assembled with full-height shroud plates |
| Degradation Effect: | See MRP-227 Table 4-5 | | |
| Primary Link: | See MRP-227 Table 4-5 | | |
| Function: | Support the weight of the core through the core support columns. | | |

**Inspection**

| | |
|---|---|
| Method: | Visual examination. See MRP-227 Table 4-5. |
| Coverage: | See MRP-227 Table 4-5 |
| Observable Effect: | The specific relevant condition is a detectable crack-like surface indication. |

**Inputs and Assumptions**

There are several inputs and assumptions that are critical to the development of acceptance criteria for the RV Internals lower core support beam weld locations. These items are stated below:

- The inspections identified in MRP-227 are intended to provide a sampling of potential locations of degradation. Under this sampling scheme, inspection of one side (surface) of the weld is intended to provide an adequate sampling for monitoring of SCC.
- The change in resistance to fracture of the RV lower core support beam welds can be correlated to the accumulated fluence at each weld location. Welds that are subject to low fluence are considered to have a high degree of resistance to fracture. Correspondingly, those welds subject to high fluence have lower resistance to fracture. This component is expected to have low fluence.
- The prediction of crack growth is based on the stress intensity factor, K, calculated using linear elastic fracture mechanics. The rate of crack growth is dependent on the amount of neutron fluence that the weld is expected to accumulate over the licensed operating lifetime. Since there has been very limited experience of SCC initiated cracks in operating PWRs to date, growth rates developed for the prediction of SCC in BWRs are assumed to be appropriate for prediction of crack growth due to SCC in PWR reactor internals.
    - For weld locations subjected to fluence levels less than or equal to $5x10^{20}$ n/cm$^2$ (E>1MeV)(0.75 dpa), the maximum of the crack growth rate in 2017 ASME Section XI Appendix C-8520 **or** the crack growth equation in ASME Code Case N-889 [52] will be used. The conditions for the use of Code Case N-889 identified in Regulatory Guide RG 1.147, Rev. 20 [67] must be met.
    - For fluence levels at or above $5x10^{20}$ n/cm$^2$ (E>1MeV)(0.75 dpa), the crack growth equation in ASME Code Case N-889 [52] will be used. The conditions for the use of Code Case N-889 identified in



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| | |
|---|---|
| **CE-ID: 5.4** | **Lower Support Structure**<br>**Lower Core Support Beams** |
| | Regulatory Guide RG 1.147, Rev. 20 [67] must be met for the development of acceptance criteria.<br>• Depending on the magnitude of cyclic loads, such as thermal transients, additional crack growth as a result of these loads may need to be considered.<br>• Acceptance criteria for indications detected using a visual exam were developed based on an assumed through-wall flaw with a length that is uniform through the wall thickness. When the critical crack length for a through wall flaw is less than that for a part through wall flaw, this assumption is reasonable and conservative, since no information is available on the flaw depth from such a visual examination.<br>• If when developing flaw acceptance criteria based on a through-wall flaw it can be shown that the average of the thermal and residual stresses across the crack face is zero, then these stresses are considered to have an insignificant contribution to the average crack growth across the through-wall crack front and the crack growth due to SCC can be predicted using only the externally applied stresses.<br>Acceptance criteria can be developed for the entire operating license of a given plant by using predicted end-of-license fluence values. However, if there are changes to these fluence projections, such as in the event of a power uprate or change in core loading pattern, it would be necessary to confirm that the inputs selected based on fluence, such as SCC growth rate and fracture toughness, remain applicable until the end of the operating license. |
| **Failure** | |
| Failure Mechanism: | Cracking (Fatigue, SCC) |
| Failure Effect: | Potential loss of core support |
| Failure Criteria: | Actively growing flaws require mitigation. Require determination of crack growth mechanism.<br>An existing flaw is unacceptable if the flaw length projected at the next inspection cycle allows a potential loss of core support. |
| **Methodology** | |
| Goal: | Demonstrate stability of the lower support structure and that the cracking mechanism will not result in growth beyond the allowable crack length over the planned inspection interval. |
| Data Requirements: | 1. Surface crack length, as determined by visual inspection:<br>a. The 2013 edition of the ASME Code Section XI: IWA-3330(a) and Figure IWA-3330-1 provide requirements for the minimum allowable separation distance that will allow multiple cracks to be treated individually. This approach is part of the guidance in this report. The requirement in Section XI is that the ligament between adjacent cracks must be greater than half of the thickness of the material. If this criterion is not met, the individual crack |



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**CE-ID: 5.4** **Lower Support Structure**

**Lower Core Support Beams**

lengths and the length of the ligament between the cracks must be summed. This total length is then compared to the allowable length. Note that the 2007 and 2010 editions of ASME Code Section XI have the equivalent content to the 2013 edition and are also applicable.

b. The flaw proximity rules in BWRVIP-158 [62] may be optionally applied in lieu of those defined in ASME Section XI: IWA-3330(a) and Figure IWA-3330-1 when appropriate.

2. Flaw Depth
   a. For one-sided visual inspections, the flaw is assumed to be through-wall.
   b. Supplemental examinations may be used to determine depths of specific flaws as the basis for a flaw-specific evaluation, if needed.
3. Confirmation that fast neutron fluence at lower core support beam is below $3x10^{20}$ n/cm$^2$ (E> 1 MeV).
4. Steady-state and applicable operating transient stresses to be used to calculate SCC and fatigue crack growth rates.
   a. When a flaw is assumed to be through-wall, secondary weld residual and local thermal stresses do not need to be considered in determination of average crack growth rates (see assumptions).
   b. For a flaw-specific part-through flaw evaluation when the flaw depth is known, secondary weld residual and thermal stresses need to be considered in determination of local circumferential and through-wall crack growth rates.
5. Limiting transient stresses to be used to calculate allowable flaw lengths.
   a. More detailed load-deformation histories may be required for elastic-plastic or limit load calculations.

Analysis:

All analyses require an assumption of the SCC crack growth expected over the upcoming period of service. The methodology for analysis of a one-sided visual examination is based on an assumption of a through-wall flaw with weld residual and thermal stresses relieved. An appropriate crack growth rate model will be applied, which may be either K dependent or K independent. In order to apply the acceptance criteria to a period beyond one refueling cycle (not to exceed a 10-year re-inspection interval), follow-up action is required to verify the assumptions used in the predicted crack-growth rate. A re-inspection of the indication at a future specified outage, for example, would provide data that could be used to satisfy this verification requirement. For detailed crack growth models used, see “Inputs and Assumptions” above.

Failure of the welds is assumed to occur when unstable crack growth is initiated from the analyzed flaw. Two options are outlined for determining the limiting allowable flaw length, based on neutron dose. Analysis methods are suggested for both pre-inspection or generic analysis (Suggested Pre-Inspection Analysis) and for flaws observed in-service (Suggested Flaw Specific Analysis), where more detailed characteristics of the flaw and its location are known. In all cases,



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**Lower Core Support Beams**

a more detailed evaluation may be completed using a semi-elliptic surface flaw, but such an evaluation would require more detailed inspection by UT.

| Fluence Range ($n/cm^2$ E>1MeV) | Dose (dpa) | Suggested Pre-Inspection Analysis | Suggested Flaw Specific Analysis |
|---|---|---|---|
| $\leq 3 \times 10^{20}$ | $\leq 0.5$ | LEFM using 150 ksi$\sqrt{in}$ as fracture toughness(1) or Limit Load | Limit Load |

Note:

1. Alternate fracture toughness values may be applied and justification for the alternate basis shall be submitted to the NRC for information within one year.

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

Option 1: LEFM Analysis

a. Establish initial crack length (and depth if determined by supplemental examinations) based on inspection results.

b. The final crack dimensions are calculated by adding 10 years of crack growth under normal loading conditions.

c. Ensure that the $\Delta K$ resulting from flow-induced vibration (FIV) is below the threshold for fatigue crack growth.

d. For primary loading at normal and upset loading conditions, the final stress intensity factor (K) for the flaw size from (b) must be lower than the fracture toughness for the material by a factor of at least 2.77. For the evaluation of partial through wall flaws, the stress intensity factor due to secondary loading does not require an additional safety factor and may be directly added to the adjusted stress intensity factor for primary loading (i.e., $2.77 \cdot K_P + K_S < K_{IC}$).

e. For the governing primary loading at emergency or faulted loading conditions, the final stress intensity factor (K) for the flaw size from (b) must be lower than the fracture toughness for the material by a factor of at least 1.39. For the evaluation of partial through wall flaws, the stress intensity factor due to secondary loading does not require an additional safety factor and may be directly added to the adjusted stress intensity factor for primary loading (i.e., $1.39 \cdot K_P + K_S < K_{IC}$).



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**CE-ID: 5.4** **Lower Support Structure**

**Lower Core Support Beams**

Option 2: Limit Load Analysis (For neutron fluence < $3x10^{20}$ n/cm$^2$ at E>1 MeV or dose less than approximately 0.5 dpa)

a. Establish initial crack length (and depth if determined by supplemental examinations) based on inspection results.

b. The final crack dimensions are calculated by adding 10 years of crack growth under normal loading conditions.

c. Ensure that the ΔK resulting from flow-induced vibration (FIV) is below the threshold for fatigue crack growth.

d. Determine the bending moment (M) that can be tolerated as a function of the postulated flaw length.

e. The applied moment, increased by a factor of 1.39 (for primary loading at emergency and faulted conditions) or 2.77 (for primary loading at normal and upset conditions) must be less than the limit moment (from step d) for the flaw length determined in (b), for the flaw to be acceptable. For the evaluation of partial through wall flaws, the moment due to secondary loading does not require an additional safety factor and may be directly added to the adjusted moment for primary loading (i.e., $[2.77 \text{ or } 1.39]\cdot M_p + M_s < M_{applied}$).

Acceptance Criteria: The lower core support beam welds continue to perform their functional requirements with the projected flaw length at the end of the inspection interval.

Approach: Limit load or LEFM analysis

Additional Action(s):

1) In order to apply the TR methodology for cracks detected via one-sided visual examinations, if supplementary examinations to determine crack depth were not performed, a follow up examination must be performed no later than the next RFO to confirm the assumed crack growth rates are conservative, unless a technical justification for a longer verification interval is submitted.

2) Note that the licensee shall submit the plant-specific crack growth and fracture mechanics analysis justifying the detected flaws, including a justification of the allowable operating period before re-inspection, within one year after any inspection that detects relevant conditions as defined in Table 5-2 of MRP-227.



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| CE-ID: 6 | Core Support Barrel Assembly<br>Middle Girth Weld | | |
|---|---|---|---|
| Category: | Primary | Applicability: | All plants |
| Degradation Effect: | See MRP-227 Table 4-2 | | |
| Expansion Link: | See MRP-227 Table 4-2 | | |
| Function: | Primary core support structure | | |

**Inspection**

| | |
|---|---|
| Method: | Visual examination. See MRP-227 Table 4-2. |
| Coverage: | See MRP-227 Table 4-2 |
| Observable Effect: | The specific relevant condition is a detectable crack-like surface indication. |

**Inputs and Assumptions**

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

- The inspections identified in MRP-227 are intended to provide a sampling of potential locations of degradation. Under this approach, inspection of one side (surface) of the weld is assumed to provide an adequate sampling for monitoring of SCC and/or Irradiation Assisted Stress Corrosion Cracking (IASCC).
- The change in resistance to fracture of the RV core barrel welds can be correlated to the accumulated fluence at each weld location. Welds that are subject to low fluence are considered to have a high degree of resistance to fracture. Correspondingly, those welds subject to high fluence have lower resistance to fracture.
- It is assumed that the primary mode of crack growth to be expected in this region is stress corrosion cracking.
- The prediction of crack growth is based on the stress intensity factor, K, calculated using linear elastic fracture mechanics. The rate of crack growth is dependent on the amount of neutron fluence that the weld is expected to accumulate over the licensed operating lifetime. Since there has been very limited experience of IASCC initiated cracks in this or comparable components or welds in operating PWRs to date, growth rate models must be obtained from other sources. One source is the models developed for predicting IASCC crack growth rate (CGR) in BWR components. The other source is the models developed based on laboratory data that may include test results for both PWR and BWR representative conditions. These models were considered to be appropriate for the prediction of crack growth due to IASCC in PWR internals.
    - For weld locations subjected to fluence levels less than or equal to $5x10^{20}$ n/cm$^2$ (E>1MeV)(0.75 dpa), the maximum of the crack growth rate in 2017 ASME Section XI Appendix C-8520 **or** the crack growth equation in ASME Code Case N-889 [52] will be



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**CE-ID: 6** **Core Support Barrel Assembly**

**Middle Girth Weld**

used. The conditions for the use of Code Case N-889 identified in Regulatory Guide RG 1.147, Rev. 20 [67] must be met.

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

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

**Failure**

| | |
|---|---|
| Failure Mechanism: | Cracking (SCC, IASCC), Aging Management (IE) |
| Failure Effect: | Potential loss-of-core support. |
| Failure Criteria: | Actively growing through-wall flaws may require mitigation.<br>An existing flaw is unacceptable if the flaw length projected at the next inspection cycle results in a potential loss-of-core support. |

**Methodology**

| | |
|---|---|
| Goal: | Demonstrate that the cracking mechanism will not result in growth beyond the allowable crack length over the planned inspection interval. |
| Data Requirements: | 1. Surface crack length as determined by visual inspection<br>a. The 2013 edition of the ASME Code Section XI: IWA-3330(a) and Figure IWA-3330-1 provide requirements for the minimum allowable separation distance that will serve as the basis for guidance in this report. The requirement in Section XI is that the ligament between adjacent cracks must be greater than half of the thickness of the material. If this criterion is not met, the individual |



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**Middle Girth Weld**

crack lengths and the length of the ligament between the cracks must be summed. This total length is then compared to the allowable length. Note that the 2007 and 2010 editions of ASME Code Section XI have the equivalent content to the 2013 edition and are also applicable.

b. The flaw proximity rules in BWRVIP-158 [62] may be optionally applied in lieu of those defined in ASME Section XI: IWA-3330(a) and Figure IWA-3330-1 when appropriate..

2. Flaw Depth
   a. For one-sided visual inspections, the flaw is assumed to be through-wall.
   b. Supplemental examinations may be used to determine flaw depth, for a flaw-specific criterion, if needed.
3. Estimate of fast neutron fluence at crack location.
4. Steady-state normal operating stresses to be used to calculate SCC/IASCC crack growth rates.
   a. Axial stresses based on normal mechanical loads are required for circumferential flaws.
   b. Stresses which have an insignificant net-through-wall value (average stress is near zero), such as weld residual stresses and thermal stresses due to local through-wall temperature gradients are considered to have minimal impact on the effective crack growth rates in through-wall flaws.
   c. Secondary weld residual and thermal stresses need to be considered in determination of circumferential and through-wall crack growth rates in partial through-wall flaws, whose dimensions would have to be determined with supplemental UT examinations.
5. Limiting externally applied transient stresses to be used to calculate allowable flaw lengths. Stresses arising from pressure, mechanical and thermal loads would be included in this calculation.

   More detailed load-deformation histories may be required for elastic-plastic or limit load calculations, if these calculations are necessary.

Analysis: All analyses require an assumption of the SCC/IASCC crack growth expected over the upcoming period of service. The methodology for analysis of a one-sided visual examination is based on an assumption of a through-wall flaw with weld residual and thermal stresses relieved. An appropriate crack growth rate model will be applied, which may be either K dependent or K independent. Fatigue crack growth has been assumed to be negligible. In order to apply the acceptance criteria to a period beyond one refueling cycle (not to exceed a 10-year re-inspection interval), additional action is required. The flaw evaluation will address the verification process of the predicted crack growth rate. Depending on the flaw size and knowledge of the plant conditions, a re-inspection at the next refueling outage may be required to provide data needed to operate beyond one cycle. The verification plan shall be included in the



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**CE-ID: 6** **Core Support Barrel Assembly**

**Middle Girth Weld**

evaluation which is submitted to the regulator for their information. For detailed crack growth models used, see "Inputs and Assumptions" above.

Failure of the middle girth welds is assumed to occur when unstable circumferential crack growth is initiated from the analyzed flaw. Three options are outlined for determining the limiting allowable flaw length, based on neutron dose. Analysis methods are suggested for both pre-inspection or generic analysis (Suggested Pre-Inspection Analysis) and for flaws observed in-service (Suggested Flaw Specific Analysis), where more detailed characteristics of the flaw and its location are known. In all cases, a more detailed evaluation may be completed using a semi-elliptic surface flaw, but such an evaluation would require more detailed inspection by UT.

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

Note:

1. Alternate fracture toughness values may be applied and justification for the alternate basis shall be submitted to the NRC for information within one year.
2. MRP-211 [66] provides the basis for the fracture toughness value.

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

Option 1: LEFM Analysis



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**CE-ID: 6** **Core Support Barrel Assembly**

**Middle Girth Weld**

a. Establish initial crack length (and depth if determined by supplemental examinations) based on inspection results.

b. The final crack dimensions are calculated by adding 10 years of crack growth under normal loading conditions.

c. Ensure that the ΔK resulting from flow-induced vibration (FIV) is below the threshold for fatigue crack growth.

d. For primary loading at normal and upset loading conditions, the final stress intensity factor (K) for the flaw size from (b) must be lower than the fracture toughness for the material by a factor of at least 2.77. For the evaluation of partial through wall flaws, the stress intensity factor due to secondary loading does not require an additional safety factor and may be directly added to the adjusted stress intensity factor for primary loading (i.e., $2.77 \cdot K_P + K_S < K_{IC}$).

e. For the governing primary loading at emergency or faulted loading conditions, the final stress intensity factor (K) for the flaw size from (b) must be lower than the fracture toughness for the material by a factor of at least 1.39. For the evaluation of partial through wall flaws, the stress intensity factor due to secondary loading does not require an additional safety factor and may be directly added to the adjusted stress intensity factor for primary loading (i.e., $1.39 \cdot K_P + K_S < K_{IC}$).

Option 2: EPFM Analysis (for neutron dose levels > 0.5 dpa but < 5 dpa)

a. Establish initial crack length (and depth if determined by supplemental examinations) based on inspection results.

b. The final crack dimensions are calculated by adding 10 years of crack growth under normal loading conditions.

c. Ensure that the ΔK resulting from flow-induced vibration (FIV) is below the threshold for fatigue crack growth.

d. Develop the (J/T) material curve from the material J-R curve.

e. The $J_{applied}$ curve must include a safety factor of 2.77 for primary loading at normal and upset conditions, and a factor of 1.39 for emergency and faulted conditions. For the evaluation of partial through wall flaws, the J curve due to secondary loading does not require an additional safety factor and may be directly added to the adjusted J curve for primary loading (i.e., $[2.77 \text{ or } 1.39] \cdot J_p + J_s < J_{applied}$).

f. The intersection of the material and applied (J/T) curves indicates the instability point. The flaw size at instability is determined from the $J_{applied}$ versus flaw size curve.

g. The flaw size at instability must be larger than the flaw size from (b), for the flaw to be acceptable.

The intersection of the material and applied (J/T) curves indicates the instability point. The load at instability is determined from the $J_{applied}$ versus load curve.



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**CE-ID: 6** **Core Support Barrel Assembly**

**Middle Girth Weld**

Option 3: Limit Load Analysis (For neutron fluence < $3x10^{20}$ n/cm$^2$ at E>1 MeV or dose less than approximately 0.5 dpa)

a. Establish initial crack length (and depth if determined by supplemental examinations) based on inspection results.

b. The final crack dimensions are calculated by adding 10 years of crack growth under normal loading conditions.

c. Ensure that the $\Delta K$ resulting from flow-induced vibration (FIV) is below the threshold for fatigue crack growth.

d. Determine the bending moment (M) that can be tolerated as a function of the postulated flaw length.

e. The applied moment, increased by a factor of 1.39 (for primary loading at emergency and faulted conditions) or 2.77 (for primary loading at normal and upset conditions) must be less than the limit moment (from step d) for the flaw length determined in (b), for the flaw to be acceptable. For the evaluation of partial through wall flaws, the moment due to secondary loading does not require an additional safety factor and may be directly added to the adjusted moment for primary loading (i.e., $[2.77 \text{ or } 1.39] \cdot M_p + M_s < M_{applied}$).

Acceptance Criteria: The middle girth welds continue to perform their functional requirements with the projected flaw length at the end of the inspection interval.

Approach: Limit load, EPFM, or LEFM analysis

Additional Action(s):

1) In order to apply the TR methodology for cracks detected via one-sided visual examinations, if supplementary examinations to determine crack depth were not performed, a follow up examination must be performed no later than the next RFO to confirm the assumed crack growth rates are conservative, unless a technical justification for a longer verification interval is submitted.

2) Note that the licensee shall submit the plant-specific crack growth and fracture mechanics analysis justifying the detected flaws, including a justification of the allowable operating period before re-inspection, within one year after any inspection that triggers the expansion criteria as defined in Table 5-2 of MRP-227.



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| CE-ID: 6.1 | Core Support Barrel Assembly<br>Middle Axial Weld | | |
|---|---|---|---|
| Category: | Expansion | Applicability: | All plants |
| Degradation Effect: | See MRP-227 Table 4-5 | | |
| Primary Link: | See MRP-227 Table 4-5 | | |
| Function: | The middle axial weld contributes to the overall structural integrity of the core barrel. | | |

**Inspection**

| | |
|---|---|
| Method: | Visual examination. See MRP-227 Table 4-5. |
| Coverage: | See MRP-227 Table 4-5. |
| Observable Effect: | The specific relevant condition is a detectable crack-like surface indication. |

**Inputs and Assumptions**

There are several inputs and assumptions that are critical to the development of acceptance criteria for the RV internals middle axial weld. These items are stated below:

- The inspections identified in MRP-227 are intended to provide a sampling of potential locations of degradation. Under this approach, inspection of one side (surface) of the weld is assumed to provide an adequate sampling for monitoring of SCC.
- The change in resistance to fracture of the RV middle axial weld can be correlated to the accumulated fluence at each weld location. Welds that are subject to low fluence are considered to have a high degree of resistance to fracture. Correspondingly, those welds subject to high fluence have lower resistance to fracture.
- It is assumed that the primary mode of crack growth to be expected in this region is stress corrosion cracking.
- The prediction of crack growth is based on the stress intensity factor, K, calculated using linear elastic fracture mechanics. The rate of crack growth is dependent on the amount of neutron fluence that the weld is expected to accumulate over the licensed operating lifetime. Since there has been very limited experience of IASCC initiated cracks in this or comparable components or welds in operating PWRs to date, growth rate models must be obtained from other sources. One source is the models developed for predicting IASCC crack growth rate (CGR) in BWR components. The other source is the models developed based on laboratory data that may include test results for both PWR and BWR representative conditions. These models were considered to be appropriate for the prediction of crack growth due to IASCC in PWR internals.
    - For weld locations subjected to fluence levels less than or equal to $5x10^{20}$ n/cm$^2$ (E>1MeV)(0.75 dpa), the maximum of the crack growth rate in 2017 ASME Section XI Appendix C-8520 **or** the crack growth equation in ASME Code Case N-889 [52] will be



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**CE-ID: 6.1** **Core Support Barrel Assembly**

**Middle Axial Weld**

used. The conditions for the use of Code Case N-889 identified in Regulatory Guide RG 1.147, Rev. 20 [67] must be met.

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

- Depending on the magnitude of cyclic loads, such as thermal transients, additional crack growth as a result of these loads may need to be considered.

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

**Failure**

| | |
|---|---|
| Failure Mechanism: | Cracking (SCC) |
| Failure Effect: | Potential for stress redistribution within the core barrel, which could initiate or accelerate circumferential crack growth in the core barrel. |
| Failure Criteria: | Actively growing through-wall flaws may require mitigation.<br>An existing flaw is unacceptable if the flaw length projected at the next inspection cycle results in a potential loss-of-core support. |

**Methodology**

| | |
|---|---|
| Goal: | Demonstrate that the cracking mechanism will not result in growth beyond the allowable crack length over the planned inspection interval. |
| Data Requirements: | 1. Surface crack length as determined by visual inspection<br>a. The 2013 edition of the ASME Code Section XI: IWA-3330(a) and Figure IWA-3330-1 provide requirements for the minimum allowable separation distance that will serve as the basis for guidance in this report. The requirement in Section XI is that the ligament between adjacent cracks must be greater than half of the thickness of the material. If this criterion is not met, the individual |



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**CE-ID: 6.1** **Core Support Barrel Assembly**

**Middle Axial Weld**

crack lengths and the length of the ligament between the cracks must be summed. This total length is then compared to the allowable length. Note that the 2007 and 2010 editions of ASME Code Section XI have the equivalent content to the 2013 edition and are also applicable.

b. The flaw proximity rules in BWRVIP-158 [62] may be optionally applied in lieu of those defined in ASME Section XI: IWA-3330(a) and Figure IWA-3330-1 when appropriate.

2. Flaw Depth
   a. For one-sided visual inspections, the flaw is assumed to be through-wall.
   b. Supplemental examinations may be used to determine flaw depth, for a flaw-specific criterion, if needed.
3. Estimate of fast neutron fluence at crack location.
4. Steady-state normal operating stresses to be used to calculate SCC crack growth rates.
   a. Hoop stresses based on normal mechanical loads are required for axial flaws.
   b. Stresses which have an insignificant net-through-wall value (average stress is near zero), such as weld residual stresses and thermal stresses due to local through-wall temperature gradients are considered to have minimal impact on the effective crack growth rates in through-wall flaws.
   c. Secondary weld residual and thermal stresses need to be considered in determination of axial and through-wall crack growth rates in partial through-wall flaws, whose dimensions would have to be determined with supplemental UT examinations.
5. Limiting externally applied transient stresses to be used to calculate allowable flaw lengths. Stresses arising from pressure, mechanical and thermal loads would be included in this calculation.
   a. More detailed load-deformation histories may be required for elastic-plastic or limit load calculations, if these calculations are necessary.

Analysis:

All analyses require an assumption of the SCC crack growth expected over the upcoming period of service. The methodology for analysis of a one-sided visual examination is based on an assumption of a through-wall flaw with weld residual and thermal stresses relieved. An appropriate crack growth rate model will be applied, which may be either K dependent or K independent. Fatigue crack growth has been assumed to be negligible. In order to apply the acceptance criteria to a period beyond one refueling cycle (not to exceed a 10-year re-inspection interval), additional action is required. The flaw evaluation will address the verification process of the predicted crack growth rate. Depending on the flaw size and knowledge of the plant conditions, a re-inspection at the next refueling outage may be required to provide data needed to operate beyond one cycle. The verification plan shall be included in the evaluation which is



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## CE-ID: 6.1 Core Support Barrel Assembly

### Middle Axial Weld

submitted to the regulator for their information. For detailed crack growth models used, see "Inputs and Assumptions" above.

Failure of the axial welds is assumed to occur when unstable axial crack growth is initiated from the analyzed flaw. Three options are outlined for determining the limiting allowable flaw length, based on neutron dose. Analysis methods are suggested for both pre-inspection or generic analysis (Suggested Pre-Inspection Analysis) and for flaws observed in-service (Suggested Flaw Specific Analysis), where more detailed characteristics of the flaw and its location are known. In all cases, a more detailed evaluation may be completed using a semi-elliptic surface flaw, but such an evaluation would require more detailed inspection by UT.

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

Note:

1. Alternate fracture toughness values may be applied and justification for the alternate basis shall be submitted to the NRC for information within one year.
2. MRP-211 [66] provides the basis for the fracture toughness value.

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



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**CE-ID: 6.1** **Core Support Barrel Assembly**

**Middle Axial Weld**

Option 1: LEFM Analysis

a. Establish initial crack length (and depth if determined by supplemental examinations) based on inspection results.

b. The final crack dimensions are calculated by adding 10 years of crack growth under normal loading conditions.

c. Ensure that the ΔK resulting from flow-induced vibration (FIV) is below the threshold for fatigue crack growth.

d. For primary loading at normal and upset loading conditions, the final stress intensity factor (K) for the flaw size from (b) must be lower than the fracture toughness for the material by a factor of at least 2.77. For the evaluation of partial through wall flaws, the stress intensity factor due to secondary loading does not require an additional safety factor and may be directly added to the adjusted stress intensity factor for primary loading (i.e., $2.77 \cdot K_P + K_S < K_{IC}$).

e. For the governing primary loading at emergency or faulted loading conditions, the final stress intensity factor (K) for the flaw size from (b) must be lower than the fracture toughness for the material by a factor of at least 1.39. For the evaluation of partial through wall flaws, the stress intensity factor due to secondary loading does not require an additional safety factor and may be directly added to the adjusted stress intensity factor for primary loading (i.e., $1.39 \cdot K_P + K_S < K_{IC}$).

Option 2: EPFM Analysis (for neutron dose levels > 0.5 dpa but < 5 dpa)

a. Establish initial crack length (and depth if determined by supplemental examinations) based on inspection results.

b. The final crack dimensions are calculated by adding 10 years of crack growth under normal loading conditions.

c. Ensure that the ΔK resulting from flow-induced vibration (FIV) is below the threshold for fatigue crack growth.

d. Develop the (J/T) material curve from the material J-R curve.

e. The $J_{applied}$ curve must include a safety factor of 2.77 for primary loading at normal and upset conditions, and a factor of 1.39 for primary loading at emergency and faulted conditions. For the evaluation of partial through wall flaws, the J curve due to secondary loading does not require an additional safety factor and may be directly added to the adjusted J curve for primary loading (i.e., $[2.77 \text{ or } 1.39] \cdot J_p + J_s < J_{applied}$).

f. The intersection of the material and applied (J/T) curves indicates the instability point. The flaw size at instability is determined from the $J_{applied}$ versus flaw size curve.

g. The flaw size at instability must be larger than the flaw size from (b), for the flaw to be acceptable.

The intersection of the material and applied (J/T) curves indicates the instability point. The load at instability is determined from the $J_{applied}$ versus load curve.



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**CE-ID: 6.1** **Core Support Barrel Assembly**

**Middle Axial Weld**

Option 3: Limit Load Analysis (For neutron fluence < $3x10^{20}$ n/cm$^2$ at E>1 MeV or dose less than approximately 0.5 dpa)

a. Establish initial crack length (and depth if determined by supplemental examinations) based on inspection results.

b. The final crack dimensions are calculated by adding 10 years of crack growth under normal loading conditions.

c. Ensure that the $\Delta K$ resulting from flow-induced vibration (FIV) is below the threshold for fatigue crack growth.

d. Determine the bending moment (M) that can be tolerated as a function of the postulated flaw length.

e. The applied moment, increased by a factor of 1.39 (for primary loading at emergency and faulted conditions) or 2.77 (for primary loading at normal and upset conditions) must be less than the limit moment (from step d) for the flaw length determined in (b), for the flaw to be acceptable. For the evaluation of partial through wall flaws, the moment due to secondary loading does not require an additional safety factor and may be directly added to the adjusted moment for primary loading (i.e., $[2.77 \text{ or } 1.39] \cdot M_p + M_s < M_{applied}$).

Acceptance Criteria: The middle axial welds continue to perform their functional requirements with the projected flaw lengths at the end of the inspection interval.

Approach: Limit load, EPFM, or LEFM analysis

Additional Action(s):

1) In order to apply the TR methodology for cracks detected via one-sided visual examinations, if supplementary examinations to determine crack depth were not performed, a follow up examination must be performed no later than the next RFO to confirm the assumed crack growth rates are conservative, unless a technical justification for a longer verification interval is submitted.

2) Note that the licensee shall submit the plant-specific crack growth and fracture mechanics analysis justifying the detected flaws, including a justification of the allowable operating period before re-inspection, within one year after any inspection that detects relevant conditions as defined in Table 5-2 of MRP-227.



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| CE-ID: 6.2 | Core Support Barrel Assembly<br>Lower Axial Weld | | |
|---|---|---|---|
| Category: | Expansion | Applicability: | All plants |
| Degradation Effect: | See MRP-227 Table 4-5 | | |
| Primary Link: | See MRP-227 Table 4-5 | | |
| Function: | The lower axial weld contributes to the overall structural integrity of the core barrel. | | |

**Inspection**

| | |
|---|---|
| Method: | Visual examination. See MRP-227 Table 4-5. |
| Coverage: | See MRP-227 Table 4-5. |
| Observable Effect: | The specific relevant condition is a detectable crack-like surface indication. |

**Inputs and Assumptions**

There are several inputs and assumptions that are critical to the development of acceptance criteria for the RV internals lower axial weld. These items are stated below:

- The inspections identified in MRP-227 are intended to provide a sampling of potential locations of degradation. Under this approach, inspection of one side (surface) of the weld is assumed to provide an adequate sampling for monitoring of SCC.
- The change in resistance to fracture of the RV lower axial weld can be correlated to the accumulated fluence at each weld location. Welds that are subject to low fluence are considered to have a high degree of resistance to fracture. Correspondingly, those welds subject to high fluence have lower resistance to fracture.
- It is assumed that the primary mode of crack growth to be expected in this region is stress corrosion cracking.
- The prediction of crack growth is based on the stress intensity factor, K, calculated using linear elastic fracture mechanics. The rate of crack growth is dependent on the amount of neutron fluence that the weld is expected to accumulate over the licensed operating lifetime. Since there has been very limited experience of IASCC initiated cracks in this or comparable components or welds in operating PWRs to date, growth rate models must be obtained from other sources. One source is the models developed for predicting IASCC crack growth rate (CGR) in BWR components. The other source is the models developed based on laboratory data that may include test results for both PWR and BWR representative conditions. These models were considered to be appropriate for the prediction of crack growth due to IASCC in PWR internals.
    - For weld locations subjected to fluence levels less than or equal to $5x10^{20}$ n/cm$^2$ (E>1MeV)(0.75 dpa), the maximum of the crack growth rate in 2017 ASME Section XI Appendix C-8520 **or** the crack growth equation in ASME Code Case N-889 [52] will be



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**CE-ID: 6.2** **Core Support Barrel Assembly**

**Lower Axial Weld**

used. The conditions for the use of Code Case N-889 identified in Regulatory Guide RG 1.147, Rev. 20 [67] must be met.

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

- Depending on the magnitude of cyclic loads, such as thermal transients, additional crack growth as a result of these loads may need to be considered.

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

**Failure**

| | |
|---|---|
| Failure Mechanism: | Cracking (SCC) |
| Failure Effect: | Potential for stress redistribution within the core barrel, which could initiate or accelerate circumferential crack growth in the core barrel. |
| Failure Criteria: | Actively growing through-wall flaws may require mitigation.<br>An existing flaw is unacceptable if the flaw length projected at the next inspection cycle results in a potential loss-of-core support. |

**Methodology**

| | |
|---|---|
| Goal: | Demonstrate that the cracking mechanism will not result in growth beyond the allowable crack length over the planned inspection interval. |
| Data Requirements: | 1. Surface crack length as determined by visual inspection<br>a. The 2013 edition of the ASME Code Section XI: IWA-3330(a) and Figure IWA-3330-1 provide requirements for the minimum allowable separation distance that will serve as the basis for guidance in this report. The requirement in Section XI is that the ligament between adjacent cracks must be greater than half of the thickness of the material. If this criterion is not met, the individual |



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**CE-ID: 6.2** **Core Support Barrel Assembly**

**Lower Axial Weld**

crack lengths and the length of the ligament between the cracks must be summed. This total length is then compared to the allowable length. Note that the 2007 and 2010 editions of ASME Code Section XI have the equivalent content to the 2013 edition and are also applicable.

b. The flaw proximity rules in BWRVIP-158 [62] may be optionally applied in lieu of those defined in ASME Section XI: IWA-3330(a) and Figure IWA-3330-1 when appropriate..

2. Flaw Depth
   a. For one-sided visual inspections, the flaw is assumed to be through-wall.
   b. Supplemental examinations may be used to determine flaw depth, for a flaw-specific criterion, if needed.
3. Estimate of fast neutron fluence at crack location.
4. Steady-state normal operating stresses to be used to calculate SCC crack growth rates.
   a. Hoop stresses based on normal mechanical loads are required for axial flaws.
   b. Stresses which have an insignificant net-through-wall value (average stress is near zero), such as weld residual stresses and thermal stresses due to local through-wall temperature gradients are considered to have minimal impact on the effective crack growth rates in through-wall flaws.
   c. Secondary weld residual and thermal stresses need to be considered in determination of axial and through-wall crack growth rates in partial through-wall flaws, whose dimensions would have to be determined with supplemental UT examinations.
5. Limiting externally applied transient stresses to be used to calculate allowable flaw lengths. Stresses arising from pressure, mechanical and thermal loads would be included in this calculation.
   a. More detailed load-deformation histories may be required for elastic-plastic or limit load calculations, if these calculations are necessary.

Analysis: All analyses require an assumption of the SCC crack growth expected over the upcoming period of service. The methodology for analysis of a one-sided visual examination is based on an assumption of a through-wall flaw with weld residual and thermal stresses relieved. An appropriate crack growth rate model will be applied, which may be either K dependent or K independent. Fatigue crack growth has been assumed to be negligible. In order to apply the acceptance criteria to a period beyond one refueling cycle (not to exceed a 10-year re-inspection interval), additional action is required. The flaw evaluation will address the verification process of the predicted crack growth rate. Depending on the flaw size and knowledge of the plant conditions, a re-inspection at the next refueling outage may be required to provide data needed to operate beyond one cycle. The verification plan shall be included in the evaluation which is



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**CE-ID: 6.2** **Core Support Barrel Assembly**

**Lower Axial Weld**

submitted to the regulator for their information. For detailed crack growth models used, see "Inputs and Assumptions" above.

Failure of the axial welds is assumed to occur when unstable axial crack growth is initiated from the analyzed flaw. Three options are outlined for determining the limiting allowable flaw length, based on neutron dose. Analysis methods are suggested for both pre-inspection or generic analysis (Suggested Pre-Inspection Analysis) and for flaws observed in-service (Suggested Flaw Specific Analysis), where more detailed characteristics of the flaw and its location are known. In all cases, a more detailed evaluation may be completed using a semi-elliptic surface flaw, but such an evaluation would require more detailed inspection by UT.

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

Note:

1. Alternate fracture toughness values may be applied and justification for the alternate basis shall be submitted to the NRC for information within one year.
2. MRP-211 [66] provides the basis for the fracture toughness value.

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



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**CE-ID: 6.2** **Core Support Barrel Assembly**

**Lower Axial Weld**

Option 1: LEFM Analysis

a. Establish initial crack length (and depth if determined by supplemental examinations) based on inspection results.

b. The final crack dimensions are calculated by adding 10 years of crack growth under normal loading conditions.

c. Ensure that the ΔK resulting from flow-induced vibration (FIV) is below the threshold for fatigue crack growth.

d. For primary loading at normal and upset loading conditions, the final stress intensity factor (K) for the flaw size from (b) must be lower than the fracture toughness for the material by a factor of at least 2.77. For the evaluation of partial through wall flaws, the stress intensity factor due to secondary loading does not require an additional safety factor and may be directly added to the adjusted stress intensity factor for primary loading (i.e., $2.77 \cdot K_P + K_S < K_{IC}$).

e. For the governing primary loading at emergency or faulted loading conditions, the final stress intensity factor (K) for the flaw size from (b) must be lower than the fracture toughness for the material by a factor of at least 1.39. For the evaluation of partial through wall flaws, the stress intensity factor due to secondary loading does not require an additional safety factor and may be directly added to the adjusted stress intensity factor for primary loading (i.e., $1.39 \cdot K_P + K_S < K_{IC}$).

Option 2: EPFM Analysis (for neutron dose levels > 0.5 dpa but < 5 dpa)

a. Establish initial crack length (and depth if determined by supplemental examinations) based on inspection results.

b. The final crack dimensions are calculated by adding 10 years of crack growth under normal loading conditions.

c. Ensure that the ΔK resulting from flow-induced vibration (FIV) is below the threshold for fatigue crack growth.

d. Develop the (J/T) material curve from the material J-R curve.

e. The $J_{applied}$ curve must include a safety factor of 2.77 for primary loading at normal and upset conditions, and a factor of 1.39 for primary loading at emergency and faulted conditions. For the evaluation of partial through wall flaws, the J curve due to secondary loading does not require an additional safety factor and may be directly added to the adjusted J curve for primary loading (i.e., $[2.77 \text{ or } 1.39] \cdot J_p + J_s < J_{applied}$).

f. The intersection of the material and applied (J/T) curves indicates the instability point. The flaw size at instability is determined from the $J_{applied}$ versus flaw size curve.

g. The flaw size at instability must be larger than the flaw size from (b), for the flaw to be acceptable.

The intersection of the material and applied (J/T) curves indicates the instability point. The load at instability is determined from the $J_{applied}$ versus load curve.



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**CE-ID: 6.2** **Core Support Barrel Assembly**

**Lower Axial Weld**

Option 3: Limit Load Analysis (For neutron fluence < $3x10^{20}$ n/cm$^2$ at E>1 MeV or dose less than approximately 0.5 dpa)

a. Establish initial crack length (and depth if determined by supplemental examinations) based on inspection results.

b. The final crack dimensions are calculated by adding 10 years of crack growth under normal loading conditions.

c. Ensure that the $\Delta K$ resulting from flow-induced vibration (FIV) is below the threshold for fatigue crack growth.

d. Determine the bending moment (M) that can be tolerated as a function of the postulated flaw length.

e. The applied moment, increased by a factor of 1.39 (for primary loading at emergency and faulted conditions) or 2.77 (for primary loading at normal and upset conditions) must be less than the limit moment (from step d) for the flaw length determined in (b), for the flaw to be acceptable. For the evaluation of partial through wall flaws, the moment due to secondary loading does not require an additional safety factor and may be directly added to the adjusted moment for primary loading (i.e., $[2.77 \text{ or } 1.39] \cdot M_p + M_s < M_{applied}$).

Acceptance Criteria: The lower axial weld continues to perform their functional requirements with the projected flaw lengths at the end of the inspection interval.

Approach: Limit load, EPFM, or LEFM analysis

Additional Action(s):

1) In order to apply the TR methodology for cracks detected via one-sided visual examinations, if supplementary examinations to determine crack depth were not performed, a follow up examination must be performed no later than the next RFO to confirm the assumed crack growth rates are conservative, unless a technical justification for a longer verification interval is submitted.

2) Note that the licensee shall submit the plant-specific crack growth and fracture mechanics analysis justifying the detected flaws, including a justification of the allowable operating period before re-inspection, within one year after any inspection that detects relevant conditions as defined in Table 5-2 of MRP-227.



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| CE-ID: 6.3 | Lower Support Structure<br>Core Support Columns |
|---|---|
| Category: | Expansion Applicability: All plants |
| Degradation Effect: | See MRP-227 Table 4-2 |
| Expansion Link: | See MRP-227 Table 4-2 |
| Function: | The support columns are a primary core support structure. The columns maintain structural integrity of the core support plate to ensure core coolable geometry and control rod insertability. |
| **Inspection** | |
| Method: | Visual examination. See MRP-227 Table 4-2. |
| Coverage: | See MRP-227 Table 4-2. |
| Observable Effect: | Fracture<br>Potential for fuel assembly misalignment |
| **Failure** | |
| Failure Mechanism: | Cracking |
| Failure Effect: | Failure of support columns will allow local deformation of core support plate.<br>Cracks initiating in welds may lead to fracture or loss of attachment to core support plate. |
| Failure Criteria: | Must establish core support column distributions required to maintain core support plate stability. (Alternative would be to demonstrate that a limited number of failures are generally acceptable.) If additional flaws are found in the lower internals assembly, additional analysis of the as-found assembly is required to determine suitability for return to service. |
| **Methodology** | |
| Goal: | Establish acceptable pattern of core support columns. |
| Data Requirements: | 1. Applicable normal/upset and emergency/faulted loads on the lower support structure.<br>2. Design information, including geometry, materials, and method of fabrication for core support plate, columns, and barrel.<br>3. Fluence accumulated by the core support columns. |



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**CE-ID: 6.3** **Lower Support Structure**

**Core Support Columns**

Analysis:

1. Establish functional requirements for core support columns:
   a. Normal/upset requirement: During the enveloping case of normal and upset service conditions operation, the system of support columns should resist core support plate deformation due to mechanical or thermal loading.
   b. Emergency/faulted requirement: During emergency and faulted service conditions loadings, the core support system must maintain structural integrity, ensure core coolable geometry, and ensure control rod insertability.
   c. Applicable stress limits shall be met in the core support columns to confirm the validity of the analytical approach, consistent with the design basis.
2. Support column analysis assumptions:
   a. Assume any column with a crack in the main body to have failed.
   b. Assume any column with a crack in the weld to result in a failure of the attachment.
3. Structural model of lower support structure.
   Create structural model of lower support structure that includes support columns and core support plate. The model should be capable of simulating postulated failure in individual columns, thus producing results to confirm functional requirements are met.
4. Structural model must be run for functional requirements A, B, and C.

Acceptance Criteria:

The observed number of failed columns must be less than the defined acceptable number of columns for normal/upset conditions for long-term operation and emergency/faulted conditions.

There is currently no basis for estimating column failure rates. Therefore, application of the acceptance criteria requires sufficient margin to provide allowance for additional flawed columns. The maximum allowable number of flawed columns is defined as follows:

$N$ = Number of core support columns

$N_{req}$ = Number of core support columns required to meet acceptable pattern

$Margin = N-N_{req}$

$N_f$ = Number of observed flawed columns

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

Approach:

Pattern analysis specific to the lower support structure design and loads. Note that a PWROG functionality analysis has been completed.

Additional Action(s):

The licensee shall compare the inspection results to functionality analysis that has been performed for these components. If the inspection findings are outside of the limits of applicability of the functionality analysis, the licensee shall submit the plant-specific analysis for the acceptable distribution of intact components within one year after any inspection that detects relevant conditions as defined in Table 5-2 of MRP-227.



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

**Inspection**

| | |
|---|---|
| Method: | See MRP-227 Table 4-2. |
| Coverage: | See MRP-227 Table 4-2 |
| Observable Effect: | The specific relevant condition is a detectable crack-like indication. |

**Inputs and Assumptions**

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

- The inspections identified in MRP-227 are intended to provide a sampling of potential locations of degradation. Under this approach, inspection of one side (surface) of the weld is intended to provide an adequate sampling for monitoring of fatigue.
- The change in resistance to fracture of the RV core barrel welds can be correlated to the accumulated fluence at each weld location. Welds that are subject to low fluence are considered to have a high degree of resistance to fracture. Correspondingly, those welds subject to high fluence have lower resistance to fracture.
- The prediction of crack growth is based on the stress intensity factor, K, calculated using linear elastic fracture mechanics. The rate of crack growth is dependent on the amount of neutron fluence that the weld is expected to accumulate over the licensed operating lifetime. Since there has been very limited experience of SCC initiated cracks in operating PWRs to date, growth rates developed for the prediction of SCC in BWRs are assumed to be appropriate for prediction of crack growth due to SCC in PWR reactor internals.
    - For weld locations subjected to fluence levels less than or equal to $5x10^{20}$ n/cm$^2$ (E>1MeV)(0.75 dpa), the maximum of the crack growth rate in 2017 ASME Section XI Appendix C-8520 **or** the crack growth equation in ASME Code Case N-889 [52] will be used. The conditions for the use of Code Case N-889 identified in Regulatory Guide RG 1.147, Rev. 20 [67] must be met.
    - For fluence levels at or above $5x10^{20}$ n/cm$^2$ (E>1MeV)(0.75 dpa), the crack growth equation in ASME Code Case N-889 [52] will be used. The conditions for the use of Code Case N-889 identified in Regulatory Guide RG 1.147, Rev. 20 [67] must be met for the development of acceptance criteria.



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**CE-ID: 7** **Core Support Barrel Assembly**

**CSB Flexure Weld**

- Additional crack growth as a result of cyclic loads, such as thermal transients, may need to be considered.
- Acceptance criteria for indications detected using a visual exam can be developed based on an assumed through-wall flaw with a length that is uniform through the wall thickness. When the critical crack length for a through wall flaw is less than that for a part through wall flaw, this assumption is reasonable and conservative, since no information is available on the flaw depth from such a visual examination.

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

**Failure**

| | |
|---|---|
| Failure Mechanism: | SCC, Fatigue |
| Failure Effect: | Loss of core support |
| Failure Criteria: | An existing flaw is present that is projected to exceed an allowable length which would cause violation of the functional requirements of the lower support structure prior to the next inspection. |

**Methodology**

Goal:

1. Demonstrate that observed flaws will not exceed the maximum allowable crack length for limiting loads that would occur under the current licensing basis (CLB) before the next inspection interval is reached.
2. For pre-inspection analysis, assume a flaw is present at the most limiting location in the weld. For evaluation of a discovered flaw, evaluated a flaw at the location and orientation of the existing flaw.

**Fracture Mechanics Evaluation Methodology for EVT-1 Inspection Acceptance Criteria:**

Data Requirements:

1. Surface crack length determined by visual inspection
   a. The 2013 edition of the ASME Code Section XI: IWA-3330(a) and Figure IWA-3330-1 provide requirements for the minimum allowable separation distance that will serve as the basis for guidance in this report. The requirement in Section XI is that the ligament between adjacent cracks must be greater than half of the thickness of the material. If this criterion is not met, the individual crack lengths and the length of the ligament between the cracks must be summed. This total length is then compared to the allowable length. Note that the 2007 and 2010 editions of ASME Code Section XI have the equivalent content to the 2013 edition and are also applicable.



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**CE-ID: 7** **Core Support Barrel Assembly**

**CSB Flexure Weld**

b. The flaw proximity rules in BWRVIP-158 [62] may be optionally applied in lieu of those defined in ASME Section XI: IWA-3330(a) and Figure IWA-3330-1 when appropriate.

2. Flaw Depth
   a. For one-sided visual inspections, the flaw is assumed to be through-wall.
   b. Supplemental examinations may be used to determine flaw depth.
3. Fast neutron fluence at crack location, or confirmation that fast neutron fluence at CSB flexure weld is below $3x10^{20}$ n/cm$^2$ (E> 1 MeV).
4. Steady-state and applicable operating transient stresses to be used to calculate SCC and fatigue crack growth rates.
   a. Axial stresses based on normal mechanical loads are required for circumferential flaws.
   b. Stresses which have an insignificant net-through-wall value (average stress is near zero), such as weld residual stresses and thermal stresses due to local through-wall temperature gradients are considered to have minimal impact on the effective crack growth rates in through-wall flaws.
   c. Secondary weld residual and thermal stresses need to be considered in determination of circumferential and through-wall crack growth rates in partial through-wall flaws, whose dimensions would have to be determined with supplemental UT examinations.
5. Limiting externally applied transient stresses to be used to calculate allowable flaw lengths.
   a. More detailed load-deformation histories may be required for elastic-plastic or limit load calculations.

Analysis:

All analyses require an assumption of the crack growth expected over the upcoming period of service. The methodology for analysis of a one-sided visual examination is based on an assumption of a through-wall flaw with weld residual and thermal stresses relieved. An appropriate crack growth rate model will be applied, which may be either K dependent or K independent. In order to apply the acceptance criteria to a period beyond one refueling cycle (not to exceed a 10-year re-inspection interval), follow-up action is required to verify the assumptions used in the predicted crack-growth rate. A re-inspection of the indication at a future specified outage, for example, would provide data that could be used to satisfy this verification requirement. For detailed crack growth models used, see “Inputs and Assumptions” above.

Additional fatigue crack growth as a result of cyclic loads, such as thermal transients, should be considered.

Failure of the weld is assumed to occur when unstable circumferential crack growth is initiated from the analyzed flaw. Two options are outlined for determining the limiting allowable flaw length, based on neutron dose. Analysis



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**CSB Flexure Weld**

methods are suggested for both pre-inspection or generic analysis (Suggested Pre-Inspection Analysis) and for flaws observed in-service (Suggested Flaw Specific Analysis), where more detailed characteristics of the flaw and its location are known. In all cases, a more detailed evaluation may be completed using a semi-elliptic surface flaw, but such an evaluation would require more detailed inspection by UT.

| Fluence Range ($n/cm^2$ E>1MeV) | Dose (dpa) | Suggested Pre-Inspection Analysis | Suggested Flaw Specific Analysis |
|---|---|---|---|
| $\leq 3 \times 10^{20}$ | $\leq 0.5$ | LEFM using 150 ksi$\sqrt{in}$ as fracture toughness(1) or Limit Load | Limit Load |

Note:

1. Alternate fracture toughness values may be applied and justification for the alternate basis shall be submitted to the NRC for information within one year.

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

Option 1: LEFM Analysis

a. Establish initial crack length (and depth if determined by supplemental examinations) based on inspection results.
b. The final crack dimensions are calculated by adding 10 years of crack growth under normal loading conditions.
c. Ensure that the $\Delta K$ resulting from flow-induced vibration (FIV) is below the threshold for fatigue crack growth.
d. For primary loading at normal and upset loading conditions, the final stress intensity factor (K) for the flaw size from (b) must be lower than the fracture toughness for the material by a factor of at least 2.77. For the evaluation of partial through wall flaws, the stress intensity factor due to secondary loading does not require an additional safety factor and may be directly added to the adjusted stress intensity factor for primary loading (i.e., $2.77 \cdot K_P + K_S < K_{IC}$).
e. For the governing primary loading at emergency or faulted loading conditions, the final stress intensity factor (K) for the flaw size from (b) must be lower than the fracture toughness for the material by a factor of at least 1.39. For the evaluation of partial through wall flaws, the stress intensity factor due to secondary loading does not require an additional



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**CE-ID: 7** **Core Support Barrel Assembly**

**CSB Flexure Weld**

safety factor and may be directly added to the adjusted stress intensity factor for primary loading (i.e., $1.39 \cdot K_P + K_S < K_{IC}$).

Option 2: Limit Load Analysis (For neutron fluence < $3x10^{20}$ n/cm$^2$ at E>1 MeV or dose less than approximately 0.5 dpa)

a. Establish initial crack length (and depth if determined by supplemental examinations) based on inspection results.

b. The final crack dimensions are calculated by adding 10 years of crack growth under normal loading conditions.

c. Ensure that the ΔK resulting from flow-induced vibration (FIV) is below the threshold for fatigue crack growth.

d. Determine the bending moment (M) that can be tolerated as a function of the postulated flaw length.

e. The applied moment, increased by a factor of 1.39 (for primary loading at emergency and faulted conditions) or 2.77 (for primary loading at normal and upset conditions) must be less than the limit moment (from step C) for the flaw length determined in (b), for the flaw to be acceptable. For the evaluation of partial through wall flaws, the moment due to secondary loading does not require an additional safety factor and may be directly added to the adjusted moment for primary loading (i.e., $[2.77 \text{ or } 1.39] \cdot M_p + M_s < M_{applied}$).

Acceptance Criteria: Projected crack growth does not violate lower support structure functional requirements.

Approach: Determine allowable flaw length using fracture mechanics evaluation.

Additional Action(s):

1) In order to apply the TR methodology for cracks detected via one-sided visual examinations, if supplementary examinations to determine crack depth were not performed, a follow up examination must be performed no later than the next RFO to confirm the assumed crack growth rates are conservative, unless a technical justification for a longer verification interval is submitted.

2) Note that the licensee shall submit the plant-specific crack growth and fracture mechanics analysis justifying the detected flaws, including a justification of the allowable operating period before re-inspection, within one year after any inspection that detects relevant conditions as defined in Table 5-2 of MRP-227.



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| | |
|---|---|
| **CE-ID: 9** | **Lower Support Structure**<br>**Core Support Plate** |
| Category: | Primary Applicability: All plants with a core support plate |
| Degradation Effect: | See MRP-227 Table 4-2 |
| Expansion Link: | See MRP-227 Table 4-2 |
| Function: | Primary core support. Provides alignment of fuel assembly. |
| **Inspection** | |
| Method: | See MRP-227 Table 4-2. |
| Coverage: | See MRP-227 Table 4-2. |
| Observable Effect: | The specific relevant condition is a detectable, crack-like surface indication. |
| **Failure** | |
| Failure Mechanism: | Aging Management (IE), Cracking (fatigue) |
| Failure Effect: | 1. Loss of core support.<br>2. Difficulty in loading fuel due to misalignment. |
| Failure Criteria: | Deflection of the core support plate beyond the maximum permissible to maintain alignment of the fuel assemblies. |
| Failure Assumptions: | • The acceptance criteria methodology assumes that cracking due to fatigue is the active degradation mechanism and that all other components in the lower internals assembly are in a nominal, nondegraded condition. If additional flaws are found in the lower internals assembly, additional analysis of the as-found assembly is required to determine suitability for return to service.<br>• Core support plate analysis assumptions:<br>1. If the flaw location is not defined, assume crack initiates at the location in the plate with highest surface tensile stress.<br>2. Assume crack propagates through the full thickness of the plate through the ligament with highest stress to the adjacent hole with highest stress. |
| **Methodology** | |
| Goal: | Cracks in the core support plate are expected to grow from hole-to-hole within the plate. The goal is to demonstrate that the cracking, which might be present, does not cause sufficient stress in other ligaments to violate functional requirements. |
| Data Requirements: | 1. Operating loads (e.g., pressure, flow, and thermal for limiting conditions).<br>2. Functional requirements for the core support plate assembly. |



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| | |
|---|---|
| **CE-ID: 9** | **Lower Support Structure**<br>**Core Support Plate** |
| Analysis: | Determine acceptable flaw pattern:<br>1. Establish functional requirements for core support plate:<br>a. During normal operation, the core support plate would have requirements for deformation.<br>b. During limiting accident transient, system must maintain structural integrity (consult design specification for functional requirements during transients).<br>2. Construct structural finite element analysis (FEA) model of lower support structure.<br>Develop a model of the lower support structure that includes the lower core support plate, support columns, and the underlying support structure attaching to the bottom of the core support barrel. The model should be capable of modeling the structural effects of a crack connecting holes in the lower core support plate (crack tip modeling not required). Sector modeling based on symmetry may be justifiable. Evaluate the displacement of the core support plate against the functional requirements.<br>3. Any single observed crack is acceptable if the resulting core support plate stresses in the FEA model meet the functional requirements for the plate, including allowable stresses and fatigue usage factor.<br>4. If unable to demonstrate acceptability of a single crack, a detailed flaw analysis is required.<br>5. Optional determination of margin for additional cracking: Repeat the evaluation for multiple cracks connecting adjacent holes. Determine the number and pattern of connected holes to violate design requirements. |
| Acceptance Criteria: | The observed cracking does not cause deflections that violate the core support plate assembly functional requirements. Fatigue analysis does not predict development of additional cracks in excess of the acceptable flaw pattern during the inspection interval. |
| Approach: | FEA (plant specific) |
| Additional Action(s): | Note that the licensee shall submit the plant-specific analysis for the acceptable distribution of intact component within one year after any inspection that detects relevant conditions as defined in Table 5-2 of MRP-227. |



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| | | | |
|---|---|---|---|
| **CE-ID: 10** | **Upper Internals Assembly**<br>**Fuel Alignment Plate** | | |
| Category: | Primary | Applicability: | All plants with core shrouds assembled with full-height shroud plates |
| Degradation Effect: | See MRP-227 Table 4-2 | | |
| Expansion Link: | See MRP-227 Table 4-2 | | |
| Function: | Provide fuel assembly alignment and support. Direct flow into upper internals. | | |
| **Inspection** | | | |
| Method: | See MRP-227 Table 4-2. | | |
| Coverage: | See MRP-227 Table 4-2. | | |
| Observable Effect: | The specific relevant condition is a detectable crack-like surface indication. | | |
| **Failure** | | | |
| Failure Mechanism: | Aging Management (IE), Cracking (fatigue) | | |
| Failure Effect: | 1. Loss of core support.<br>2. Difficulty in loading fuel due to misalignment. | | |
| Failure Criteria: | Deflection of the fuel alignment plate beyond the maximum permissible to maintain alignment of the fuel assemblies. | | |
| Failure Assumptions: | • The acceptance criteria methodology assumes that cracking due to fatigue is the active degradation mechanism and that all other components in the upper internals assembly are in a nominal, nondegraded condition. If additional flaws are found in the upper internals assembly, additional analysis of the as-found assembly is required to determine suitability for return to service.<br>• Fuel alignment plate analysis assumptions:<br>1. If the flaw location is not defined, assume the crack initiates at the hole in the plate, with the highest surface tensile stress.<br>2. Assume the crack propagates through the full thickness of the plate through the ligament with highest stress to the adjacent hole. | | |
| **Methodology** | | | |
| Goal: | Cracks in the fuel alignment plate are expected to grow from hole-to-hole within the plate. The goal is to demonstrate that the cracking, which might be present, does not cause sufficient stress in other ligaments to violate functional requirements. | | |
| Data Requirements: | 1. Operating loads (e.g., pressure, flow, and thermal for limiting conditions).<br>2. Functional requirements for the upper core plate assembly. | | |



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**CE-ID: 10** **Upper Internals Assembly**

**Fuel Alignment Plate**

Analysis: Determine acceptable flaw pattern:

1. Establish functional requirements for the fuel assembly alignment plate:
   a. During normal operation (provide fuel assembly alignment and support and direct flow into the upper internals).
   b. During limiting accident transient, the system must maintain structural integrity (consult design specification for functional requirements during transients).
2. Construct structural finite element analysis (FEA) model of upper guide structure.
   Develop a model of the upper guide structure that includes the control element assembly (CEA) tubes and fuel alignment plate (FAP). The model should be capable of modeling the structural effects of a crack connecting holes in the lower core support plate (crack tip modeling not required). Sector modeling based on symmetry may be justifiable.
3. Any single observed crack is acceptable if the resulting stresses in the FEA model meet the functional requirements, including allowable stresses and fatigue usage factor.
4. If unable to demonstrate acceptability of a single crack, a detailed flaw analysis is required
5. Optional determination of margin for additional cracking: Repeat evaluation for multiple cracks connecting adjacent holes. Determine the number and pattern of connected holes, which causes violation of the functional requirements.

Acceptance Criteria: The observed cracking does not cause deflections that violate the fuel alignment plate assembly functional requirements. Fatigue analysis does not predict development of additional cracks in excess of the acceptable flaw pattern during the inspection interval.

Approach: FEA (plant specific)

Additional Action(s): Note that the licensee shall submit the plant-specific analysis for this component within one year after any inspection that detects relevant conditions as defined in Table 5-2 of MRP-227.



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| CE-ID: 11 | Control Element Assembly<br>Instrument Guide Tubes | | |
|---|---|---|---|
| Category: | Primary | Applicability: | All plants with instrument guide tubes in the CEA shroud assembly. |
| Degradation Effect: | See MRP-227 Table 4-2 | | |
| Expansion Link: | See MRP-227 Table 4-2 | | |
| Function: | Define path for insertion of in-core instrumentation. | | |

**Inspection**

| | |
|---|---|
| Method: | Visual examination. See MRP-227 Table 4-2.<br>Plant-specific component integrity assessments may be required if degradation is detected and remedial action is needed. |
| Coverage: | See MRP-227 Table 4-2 |
| Observable Effect: | Missing or broken supports. |

**Failure**

| | |
|---|---|
| Failure Mechanism: | Cracking |
| Failure Effect: | 1. Potential loose parts<br>2. Inability to insert/withdraw instrumentation |
| Failure Criteria: | 1. Potential uncontained loose parts<br>2. Inability to maintain minimum in-core instrumentation |

**Methodology**

| | |
|---|---|
| Goal: | Demonstrate ability to insert instrumentation. |
| Data Requirements: | Instrumentation requirements for plant. |
| Analysis: | 1. Evaluate stability of failed instrument guide tube. Any section that could potentially detach and become a loose part or otherwise interfere with plant operation should be removed or stabilized.<br>2. Any instrument guide tube with an observable crack will be assumed to have failed. |
| Acceptance Criteria: | 1. Configuration of unfailed guide tubes should be sufficient to allow adequate core monitoring.<br>2. No margin is required for this item. If the instrumentation is functional at start-up, the plant can be operated. |
| Approach: | Pass/Fail inspection with established minimum number of instrumentation tubes. Based directly on plant specifications. |



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| | |
|---|---|
| **CE-ID: 11** | **Control Element Assembly Instrument Guide Tubes** |
| Additional Action(s): | None |



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| CE-ID: 11.1 | Control Element Assembly<br>Remaining Instrument Guide Tubes | | |
|---|---|---|---|
| Category: | Expansion | Applicability: | All plants with instrument guide tubes in the CEA shroud assembly |
| Degradation Effect: | See MRP-227 Table 4-5 | | |
| Primary Link: | See MRP-227 Table 4-5 | | |
| Function: | Define path for insertion of in-core instrumentation. | | |

**Inspection**

| | |
|---|---|
| Method: | Visual examination. See MRP-227 Table 4-5. |
| Coverage: | See MRP-227 Table 4-5. |
| Observable Effect: | Missing or broken supports |

**Failure**

| | |
|---|---|
| Failure Mechanism: | Cracking |
| Failure Effect: | 1. Potential loose parts<br>2. Inability to insert/withdraw instrumentation |
| Failure Criteria: | 1. Potential uncontained loose parts<br>2. Inability to maintain minimum in-core instrumentation |

**Methodology**

| | |
|---|---|
| Goal: | Demonstrate ability to insert instrumentation. |
| Data Requirements: | Instrumentation requirements for plant. |
| Analysis: | 1. Evaluate stability of failed instrument guide tube. Any section that could potentially detach and become a loose part or otherwise interfere with plant operation should be removed or stabilized.<br>2. Any instrument guide tube with an observable crack will be assumed to have failed. |
| Acceptance Criteria: | 1. Configuration of unfailed guide tubes should be sufficient to allow adequate core monitoring.<br>2. No margin is required for this item. If the instrumentation is functional at start-up, the plant can be operated. |
| Approach: | Pass/Fail inspection with established minimum number of instrumentation tubes. Based directly on plant specifications. |
| Additional Action(s): | None |



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| | | | |
|---|---|---|---|
| **CE-ID: 12** | **Lower Support Structure**<br>**Deep Beams** | | |
| Category: | Primary | Applicability: | All plants with core shrouds assembled with full-height shroud plates |
| Degradation Effect: | See MRP-227 Table 4-2 | | |
| Expansion Link: | See MRP-227 Table 4-2 | | |
| Function: | Primary core support structure | | |

**Inspection**

| | |
|---|---|
| Method: | Visual examination. See MRP-227 Table 4-2. |
| Coverage: | See MRP-227 Table 4-2. |
| Observable Effect: | Fatigue crack growth along beams at welds. Check for a detectable surface-breaking indication in the welds or beams. |

**Inputs and Assumptions**

There are several inputs and assumptions that are critical to the development of acceptance criteria for the deep beam locations. These items are stated below:

- The inspections identified in MRP-227 are intended to provide a sampling of potential locations of degradation. Under this approach, inspection of one side (surface) of the weld is intended to provide an adequate sampling for monitoring of fatigue.
- The change in resistance to fracture of the deep beam welds can be correlated to the accumulated fluence at each weld location. Welds that are subject to low fluence are considered to have a high degree of resistance to fracture. Correspondingly, those welds subject to high fluence have lower resistance to fracture.
- The prediction of crack growth is based on the stress intensity factor, K, calculated using linear elastic fracture mechanics. The rate of crack growth is dependent on the amount of neutron fluence that the weld is expected to accumulate over the licensed operating lifetime. Since there has been very limited experience of SCC initiated cracks in operating PWRs to date, growth rates must be obtained from other sources. One source is the models developed for predicting SCC crack growth rate (CGR) in BWR components. These models were considered to be appropriate for prediction of crack growth due to SCC in PWR reactor internals.
    - For weld locations subjected to fluence levels less than or equal to $5x10^{20}$ n/cm$^2$ (E>1MeV)(0.75 dpa), the maximum of the crack growth rate in 2017 ASME Section XI Appendix C-8520 **or** the crack growth equation in ASME Code Case N-889 [52] will be used. The conditions for the use of Code Case N-889 identified in Regulatory Guide RG 1.147, Rev. 20 [67] must be met.
    - For weld locations subjected to fluence levels greater than $5x10^{20}$ n/cm$^2$ (E>1MeV) (0.75 dpa) up to and including $1.4x10^{22}$ n/cm$^2$ (E>1MeV)(20 dpa), the crack growth equation in ASME Code Case N-889 [52] will be used. The conditions for the use of Code Case



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| | |
|---|---|
| **CE-ID: 12** | **Lower Support Structure**<br>**Deep Beams** |

N-889 identified in Regulatory Guide RG 1.147, Rev. 20 [67] must be met for the development of acceptance criteria.

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

- Since the degradation effect is fatigue, low and high cycle fatigue crack growth should be evaluated. Inputs to consider include the magnitude of cyclic loads, such as thermal transients.
- The evaluation must consider potential functional limits on flaw patterns due to the component geometry.

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

| **Failure** | |
|---|---|
| Failure Mechanism: | Cracking (fatigue) |
| Failure Effect: | Loss of fuel assembly alignment |
| Failure Criteria: | An existing flaw is present that is projected to exceed an allowable length, which would cause violation of the functional requirements of the lower support structure prior to the next inspection. |

**Methodology**

Goal:

1. Demonstrate that observed flaws will not exceed the maximum allowable crack length for limiting loads that would occur under the current licensing basis (CLB) before the next inspection interval is reached.
2. For pre-inspection analysis, assume a flaw is present at the most limiting location in the weld. For evaluation of a discovered flaw, evaluate a flaw at the location and orientation of the existing flaw.

**<u>Fracture Mechanics Evaluation Methodology for EVT-1 Inspection Acceptance Criteria:</u>**

Data Requirements:

1. Surface crack length determined by visual inspection
   a. The 2013 edition of the ASME Code Section XI: IWA-3330(a) and Figure IWA-3330-1 provide requirements for the minimum allowable separation distance that will serve as the basis for guidance in this report. The requirement in Section XI is that the ligament between adjacent cracks must be greater than half of the



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**CE-ID: 12** **Lower Support Structure**

**Deep Beams**

thickness of the material. If this criterion is not met, the individual crack lengths and the length of the ligament between the cracks must be summed. This total length is then compared to the allowable length. Note that the 2007 and 2010 editions of ASME Code Section XI have the equivalent content to the 2013 edition and are also applicable.

b. The flaw proximity rules in BWRVIP-158 [62] may be optionally applied in lieu of those defined in ASME Section XI: IWA-3330(a) and Figure IWA-3330-1 when appropriate.

2. Flaw Depth
   a. For one-sided visual inspections, the flaw is assumed to be through-wall.
   b. Supplemental examinations may be used to determine flaw depth.
3. Estimate of fast neutron fluence at crack location.
4. Steady-state and applicable operating transient stresses to be used to calculate SCC and fatigue crack growth rates.
   a. Horizontal stresses based on normal mechanical loads are required for vertical flaws.
   b. Secondary weld residual and thermal stresses need to be considered in determination of through-wall crack growth rates in partial through-wall flaws.
5. Limiting externally applied transient stresses to be used to calculate allowable flaw lengths.
   a. More detailed load-deformation histories may be required for elastic-plastic or limit load calculations.

Analysis:

All analyses require an assumption of the crack growth expected over the upcoming period of service. The methodology for analysis of a one-sided visual examination is based on an assumption of a through-wall flaw with weld residual and thermal stresses relieved. An appropriate crack growth rate model will be applied, which may be either K dependent or K independent. In order to apply the acceptance criteria to a period beyond one refueling cycle (not to exceed a 10-year re-inspection interval, follow-up action is required to verify the assumptions used in the predicted crack-growth rate. A re-inspection of the indication at a future specified outage, for example, would provide data that could be used to satisfy this verification requirement.

Failure of the deep beam welds is assumed to occur when unstable crack growth is initiated from the analyzed flaw. Three options are outlined for determining the limiting allowable flaw length, based on neutron dose. Analysis methods are suggested for both pre-inspection and generic analysis (Suggested Pre-Inspection Analysis) and for flaws observed in-service (Suggested Flaw Specific Analysis), where more detailed characteristics of the flaw and its location are known. In all cases, a more detailed evaluation may be completed using a semi-elliptic surface flaw, but such an evaluation would require more detailed inspection by UT.



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**CE-ID: 12** **Lower Support Structure**

**Deep Beams**

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

Note:

1. Alternate fracture toughness values may be applied and justification for the alternate basis shall be submitted to the NRC for information within one year.
2. MRP-211 [66] provides the basis for the fracture toughness value.

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

Option 1: LEFM Analysis

a. Establish initial crack length (and depth if determined by supplemental examinations) based on inspection results.
b. The final crack dimensions are calculated by adding 10 years of crack growth under normal loading conditions.
c. Ensure that the $\Delta K$ resulting from flow-induced vibration (FIV) is below the threshold for fatigue crack growth.
d. For primary loading at normal and upset loading conditions, the final stress intensity factor (K) for the flaw size from (b) must be lower than the fracture toughness for the material by a factor of at least 2.77. For the evaluation of partial through wall flaws, the stress intensity factor due to secondary loading does not require an additional safety factor and may be directly added to the adjusted stress intensity factor for primary loading (i.e., $2.77 \cdot K_P + K_S < K_{IC}$).



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**CE-ID: 12** **Lower Support Structure**

**Deep Beams**

e. For the governing primary loading at emergency or faulted loading conditions, the final stress intensity factor (K) for the flaw size from (b) must be lower than the fracture toughness for the material by a factor of at least 1.39. For the evaluation of partial through wall flaws, the stress intensity factor due to secondary loading does not require an additional safety factor and may be directly added to the adjusted stress intensity factor for primary loading (i.e., $1.39 \cdot K_P + K_S < K_{IC}$).

Option 2: EPFM Analysis (for neutron dose levels > 0.5 dpa but < 5 dpa)

a. Establish initial crack length (and depth if determined by supplemental examinations) based on inspection results.

b. The final crack dimensions are calculated by adding 10 years of crack growth under normal loading conditions.

c. Ensure that the ΔK resulting from flow-induced vibration (FIV) is below the threshold for fatigue crack growth.

d. Develop the (J/T) material curve from the material J-R curve.

e. The $J_{applied}$ curve must include a safety factor of 2.77 for primary loading at normal and upset conditions, and a factor of 1.39 for primary loading at emergency and faulted conditions. For the evaluation of partial through wall flaws, the J curve due to secondary loading does not require an additional safety factor and may be directly added to the adjusted J curve for primary loading (i.e., $[2.77 \text{ or } 1.39] \cdot J_p + J_s < J_{applied}$).

f. The intersection of the material and applied (J/T) curves indicates the instability point. The flaw size at instability is determined from the $J_{applied}$ versus flaw size curve.

g. The flaw size at instability must be larger than the flaw size from (b), for the flaw to be acceptable.

The intersection of the material and applied (J/T) curves indicates the instability point. The load at instability is determined from the $J_{applied}$ versus load curve.

Option 3: Limit Load Analysis (For neutron fluence < $3 \times 10^{20}$ n/cm$^2$ at E>1 MeV or dose less than approximately 0.5 dpa)

a. Establish initial crack length (and depth if determined by supplemental examinations) based on inspection results.

b. The final crack dimensions are calculated by adding 10 years of crack growth under normal loading conditions.

c. Ensure that the ΔK resulting from flow-induced vibration (FIV) is below the threshold for fatigue crack growth.

d. Determine the bending moment (M) that can be tolerated as a function of the postulated flaw length.

e. The applied moment, increased by a factor of 1.39 (for primary loading at emergency and faulted conditions) or 2.77 (for primary loading at normal and upset conditions) must be less than the limit moment (from step d) for



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| | |
|---|---|
| **CE-ID: 12** | **Lower Support Structure** |
| | **Deep Beams** |
| | the flaw length determined in (b), for the flaw to be acceptable. For the evaluation of partial through wall flaws, the moment due to secondary loading does not require an additional safety factor and may be directly added to the adjusted moment for primary loading (i.e., [2.77 or 1.39]·$M_p$ + $M_s$ < $M_{applied}$). |
| Acceptance Criteria: | The deep beams continue to perform their functional requirements with the projected flaw length at the end of the inspection interval. |
| Approach: | Limit load, EPFM, or LEFM analysis. |
| Additional Action(s): | None |



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# APPENDIX D
# FLOWCHARTS FOR COMBUSTION ENGINEERING-DESIGN PRIMARY AND EXPANSION COMPONENT ITEM INSPECTIONS

**CE Primary and Expansion Components**

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

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

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

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

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

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

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

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

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

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

CE-ID: 12 Lower Support Structure – Deep Beams



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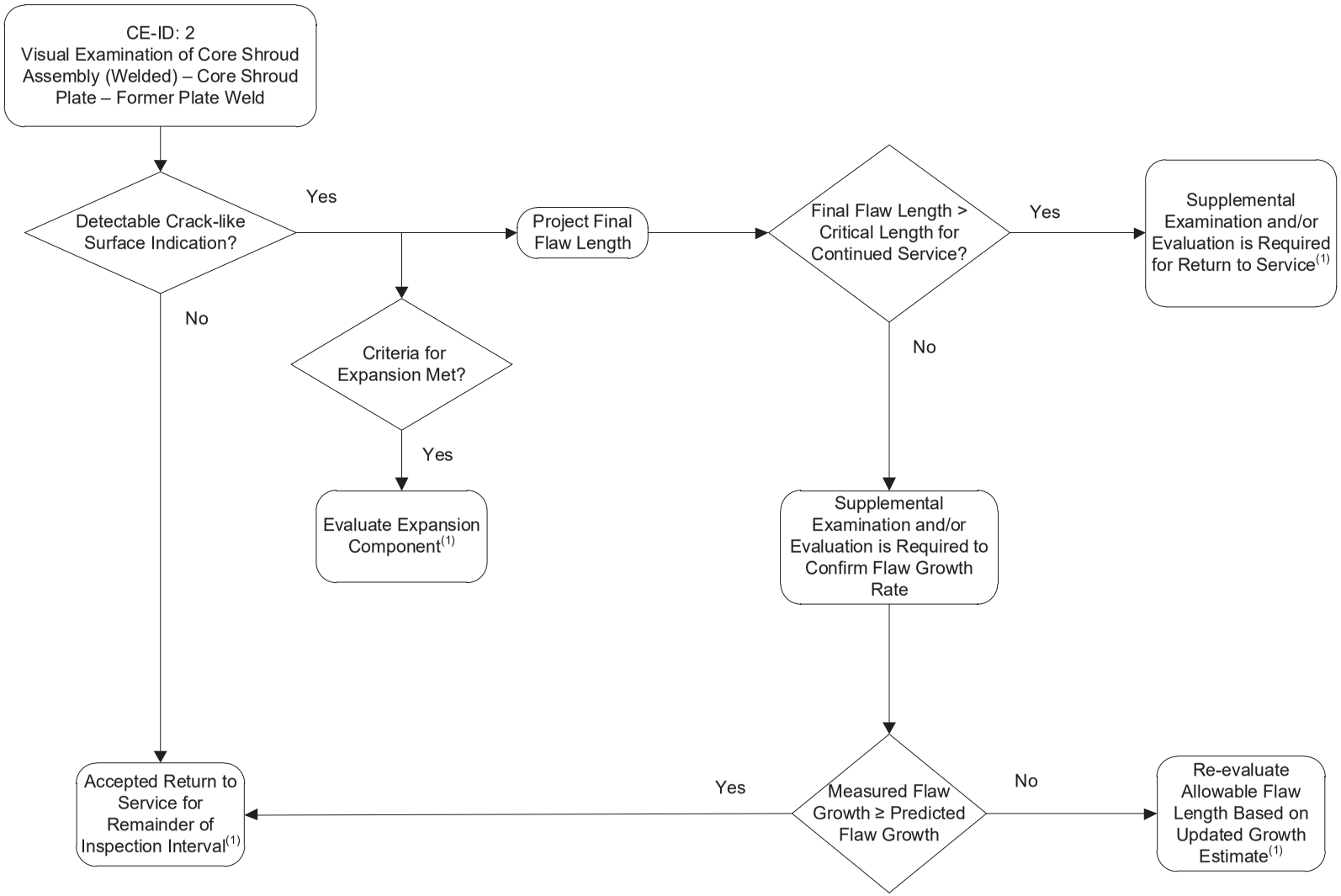
CE-ID: 2
Core Shroud Assembly (Welded) Core Shroud Plate – Former Plate Weld

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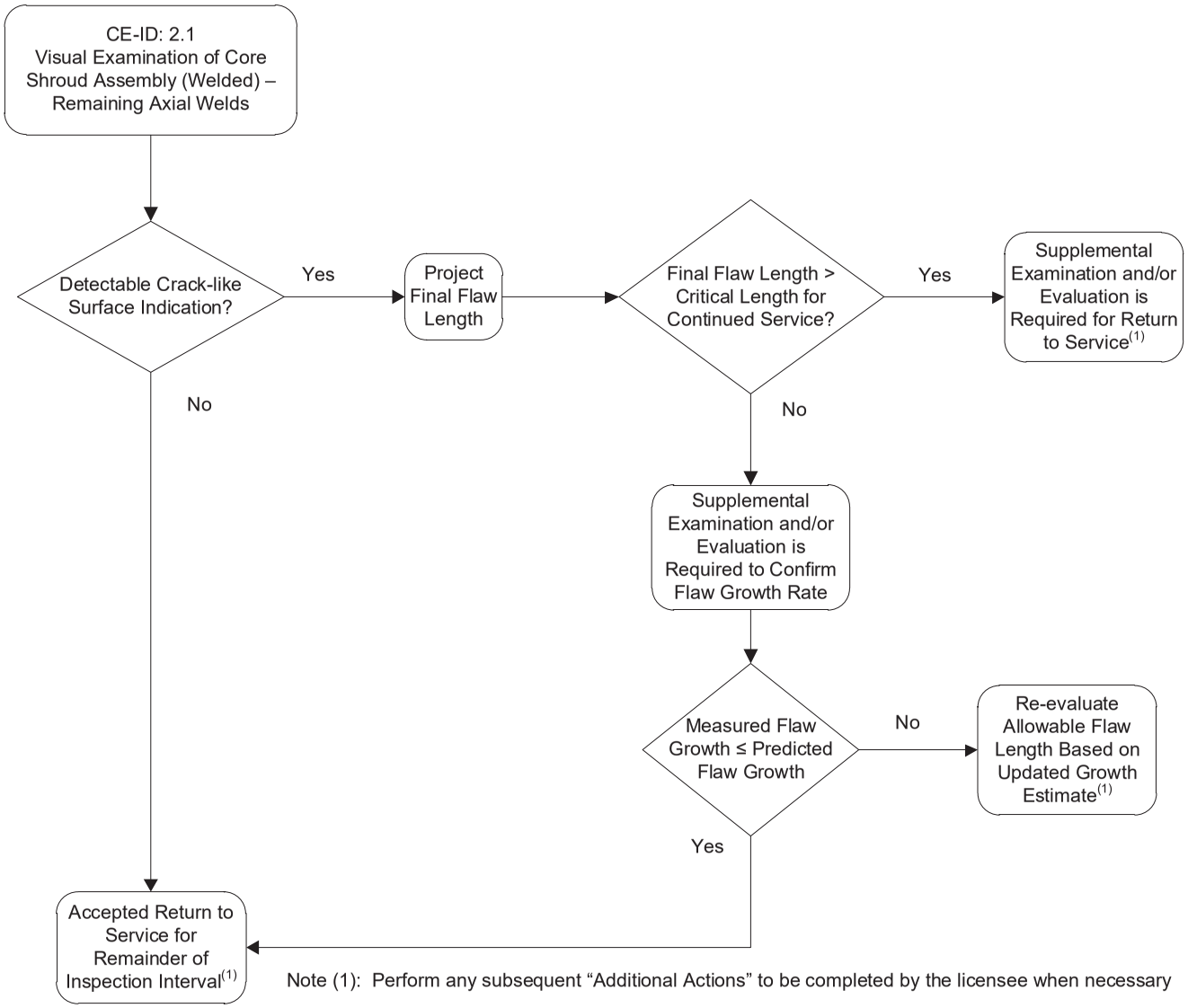
## CE-ID: 2.1
## Core Shroud Assembly (Welded) - Remaining Axial Welds



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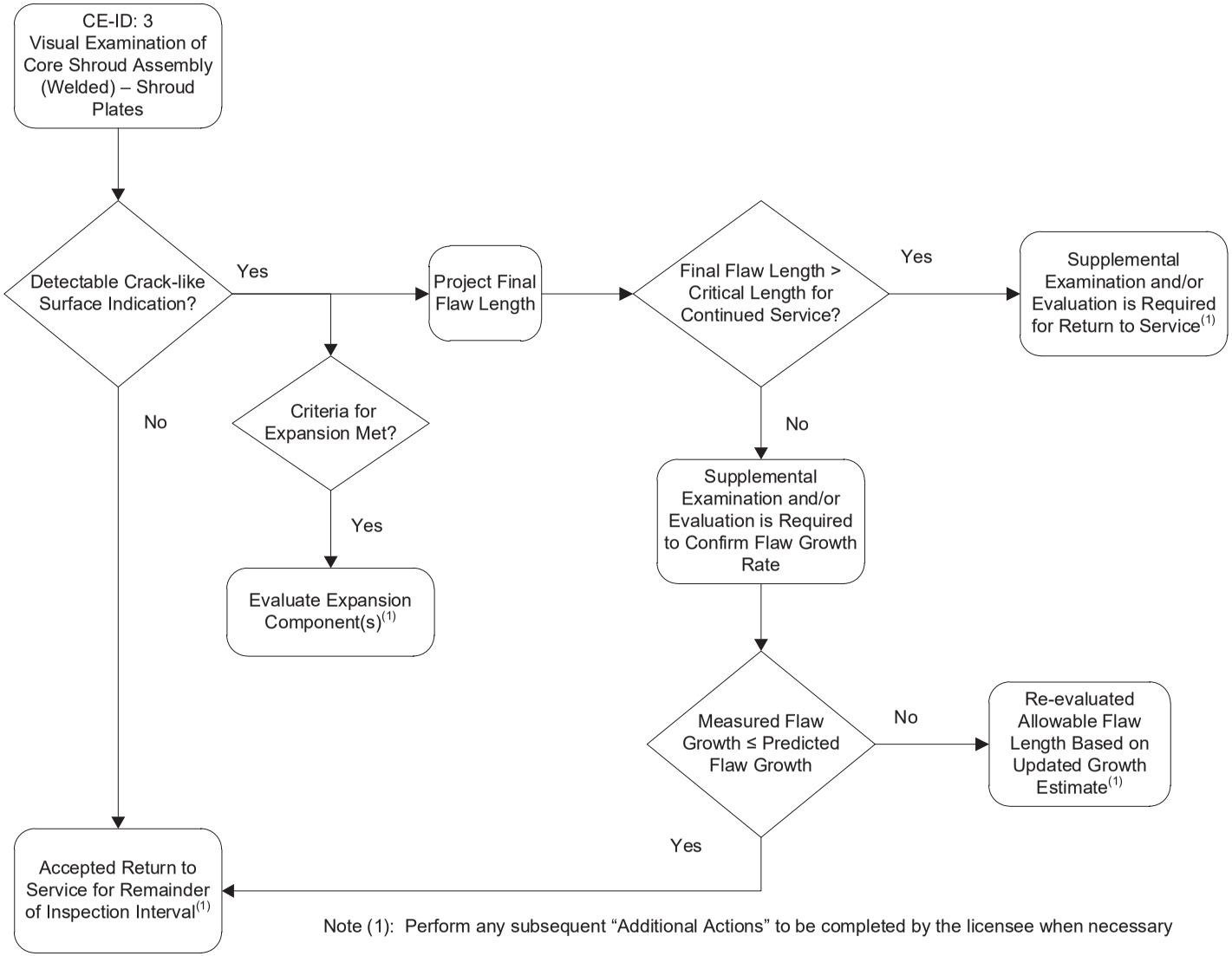
CE-ID: 3
Core Shroud Assembly (Welded) - Shroud Plates

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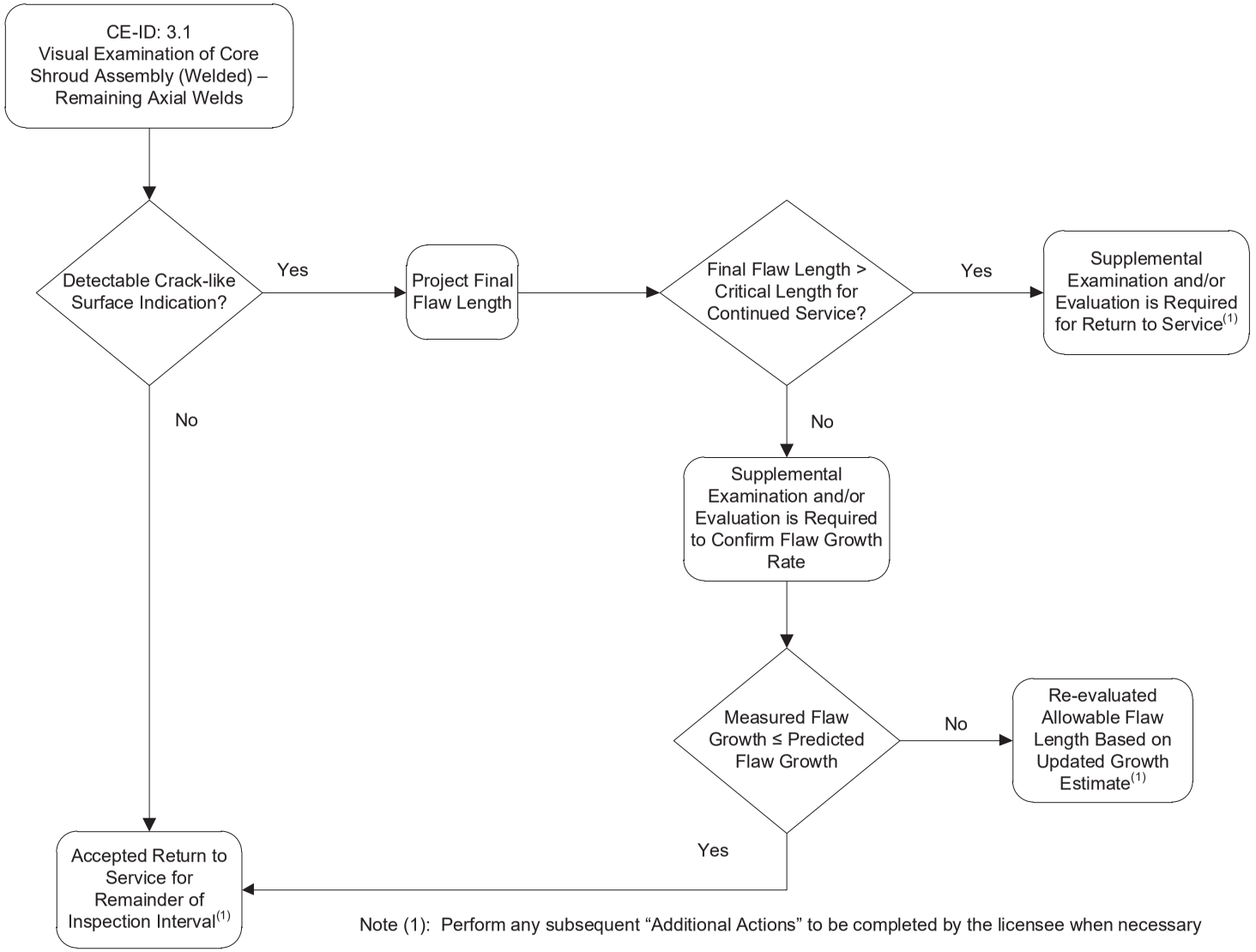
# CE-ID: 3.1
# Core Shroud Assembly (Welded) – Remaining Axial Welds

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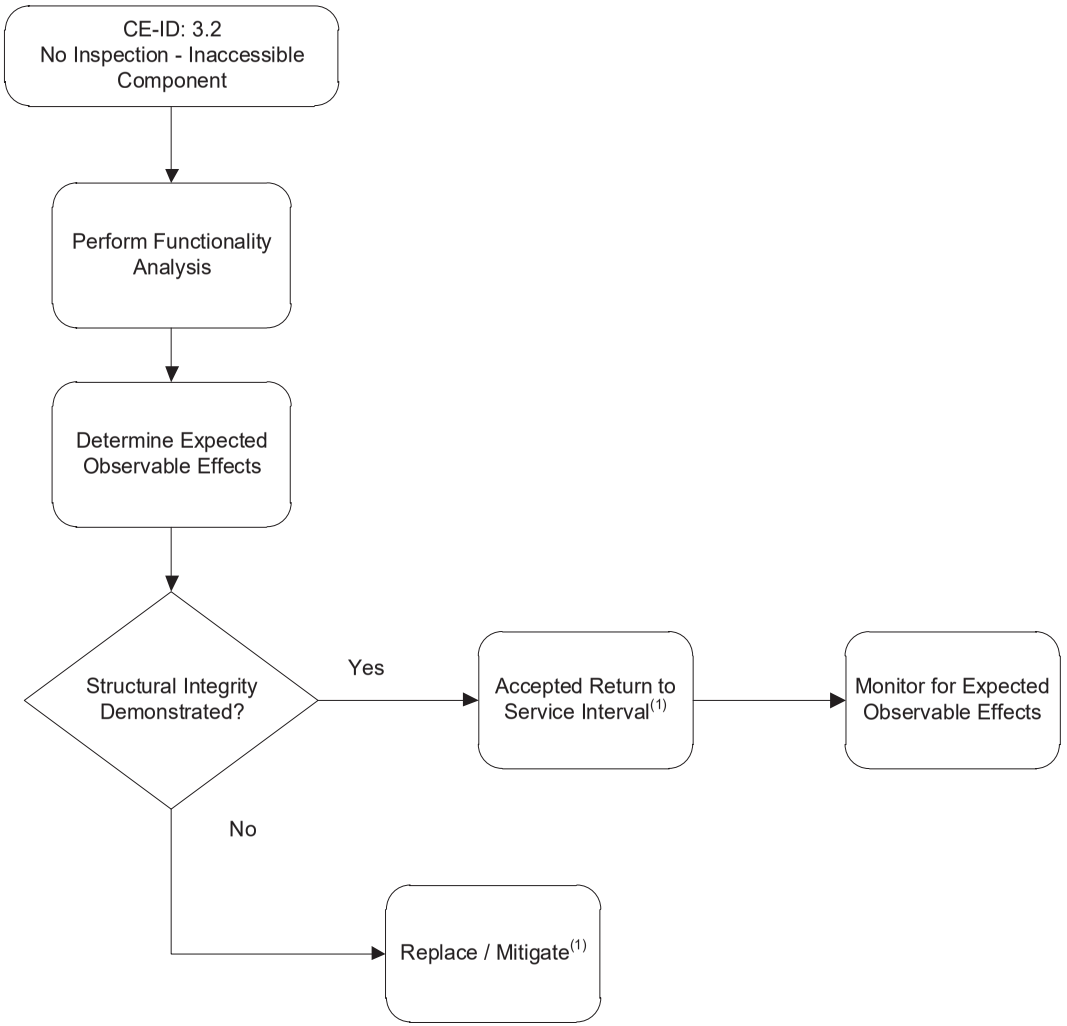
## CE-ID: 3.2
## Core Shroud Assembly (Welded) – Ribs and Rings

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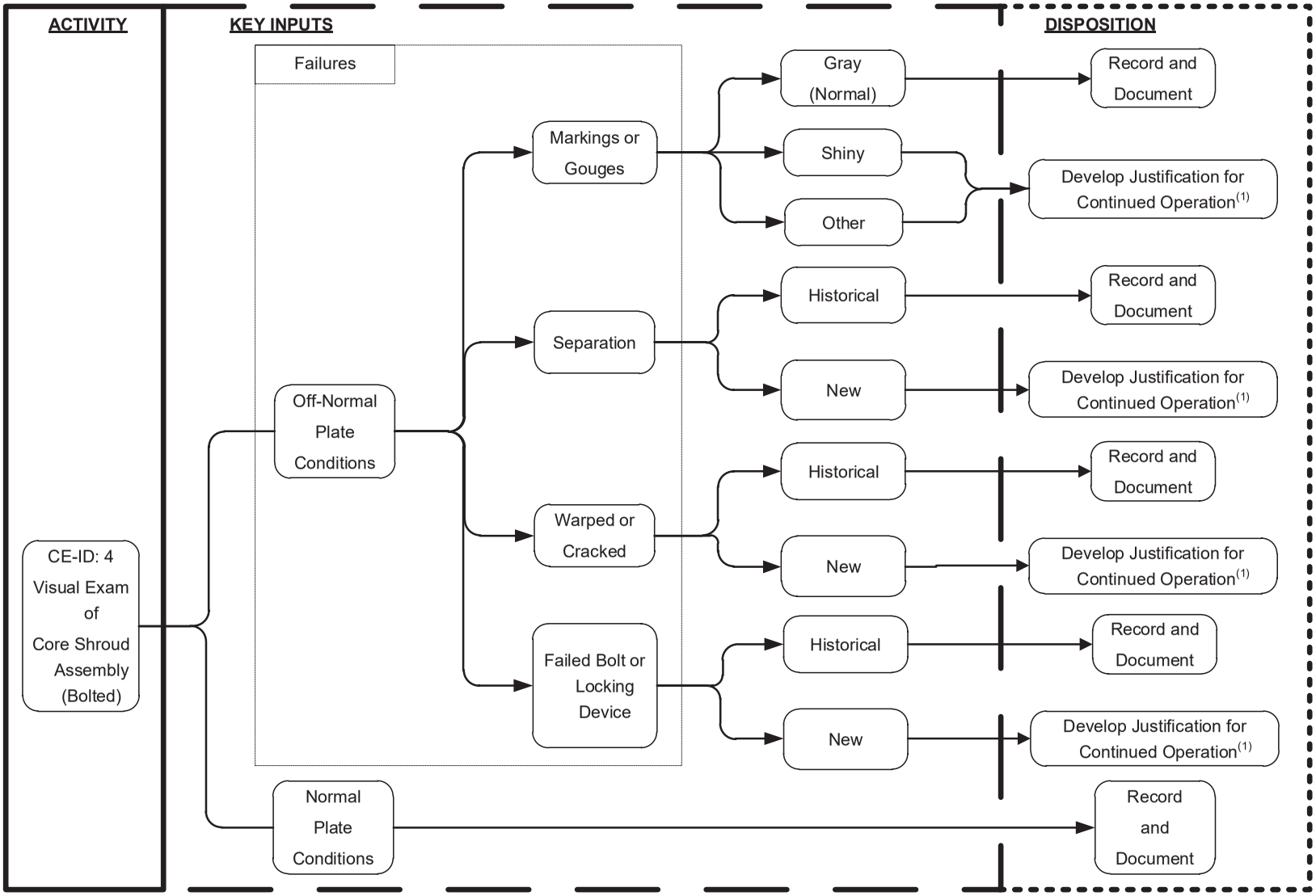
CE-ID:4
Core Shroud Assemby (Bolted) FMEA Layout (Example)

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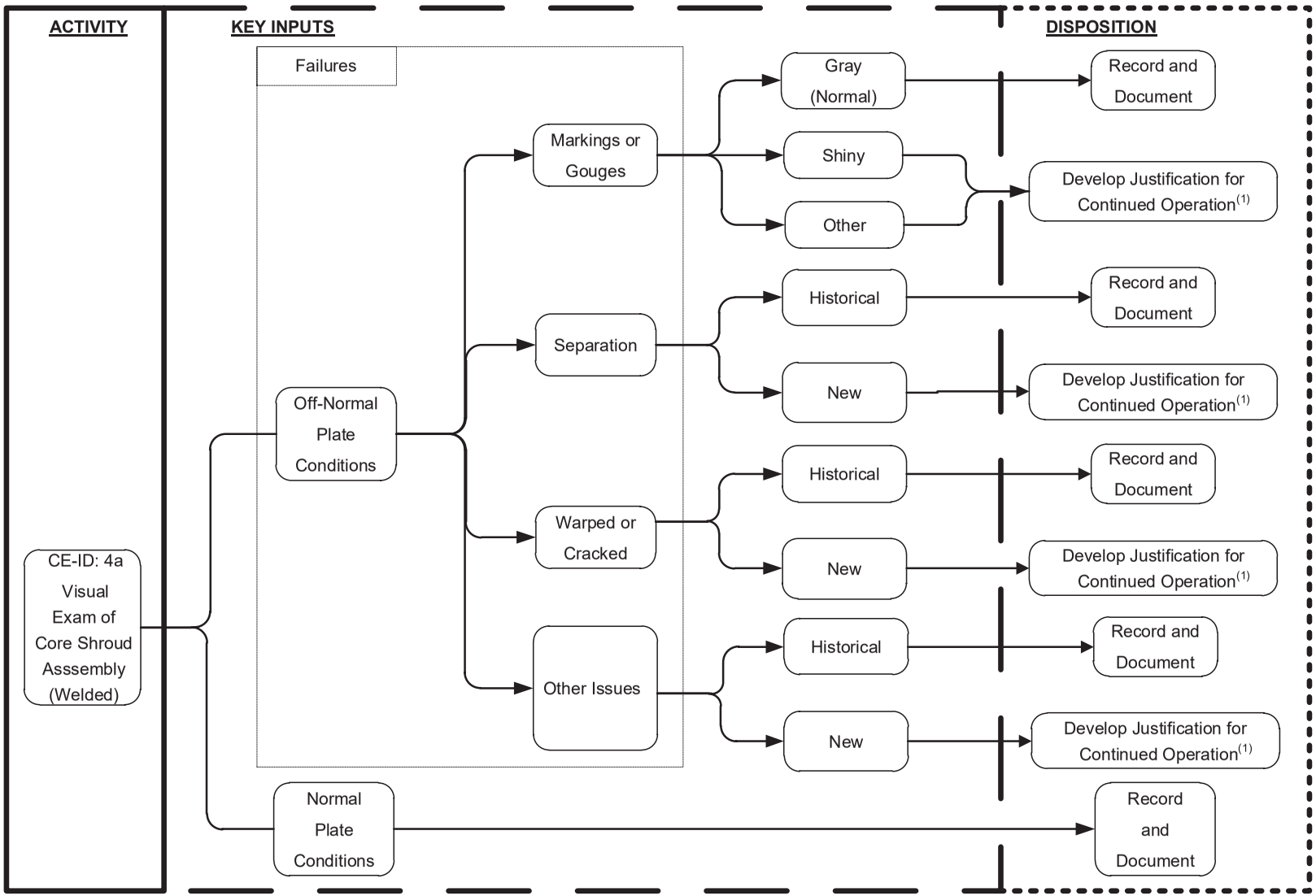
CE-ID:4a
Core Shroud Assemby (Welded) FMEA Layout (Example)

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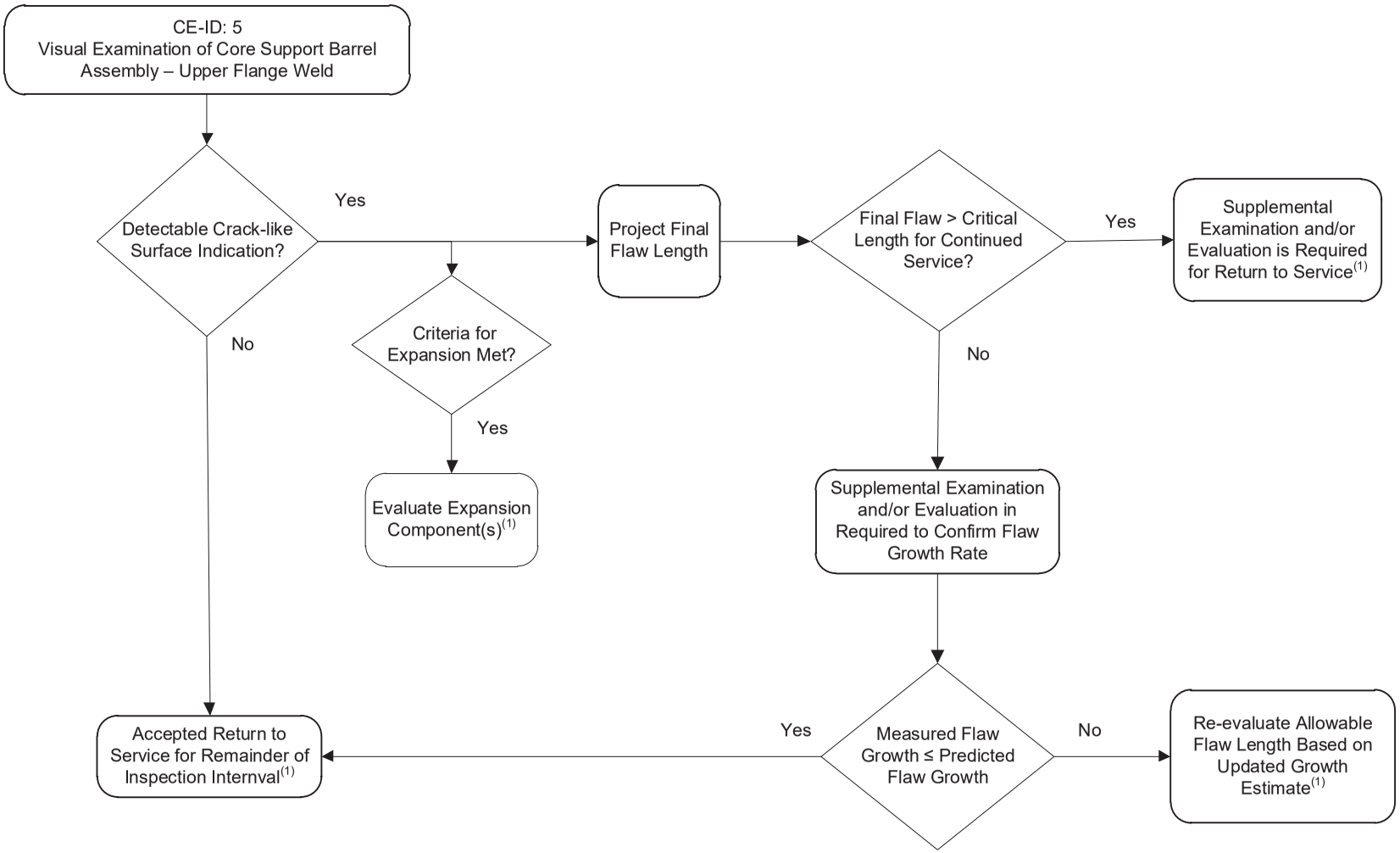
# CE-ID:5
## Core Support Barrel Assembly - Upper Flange Weld

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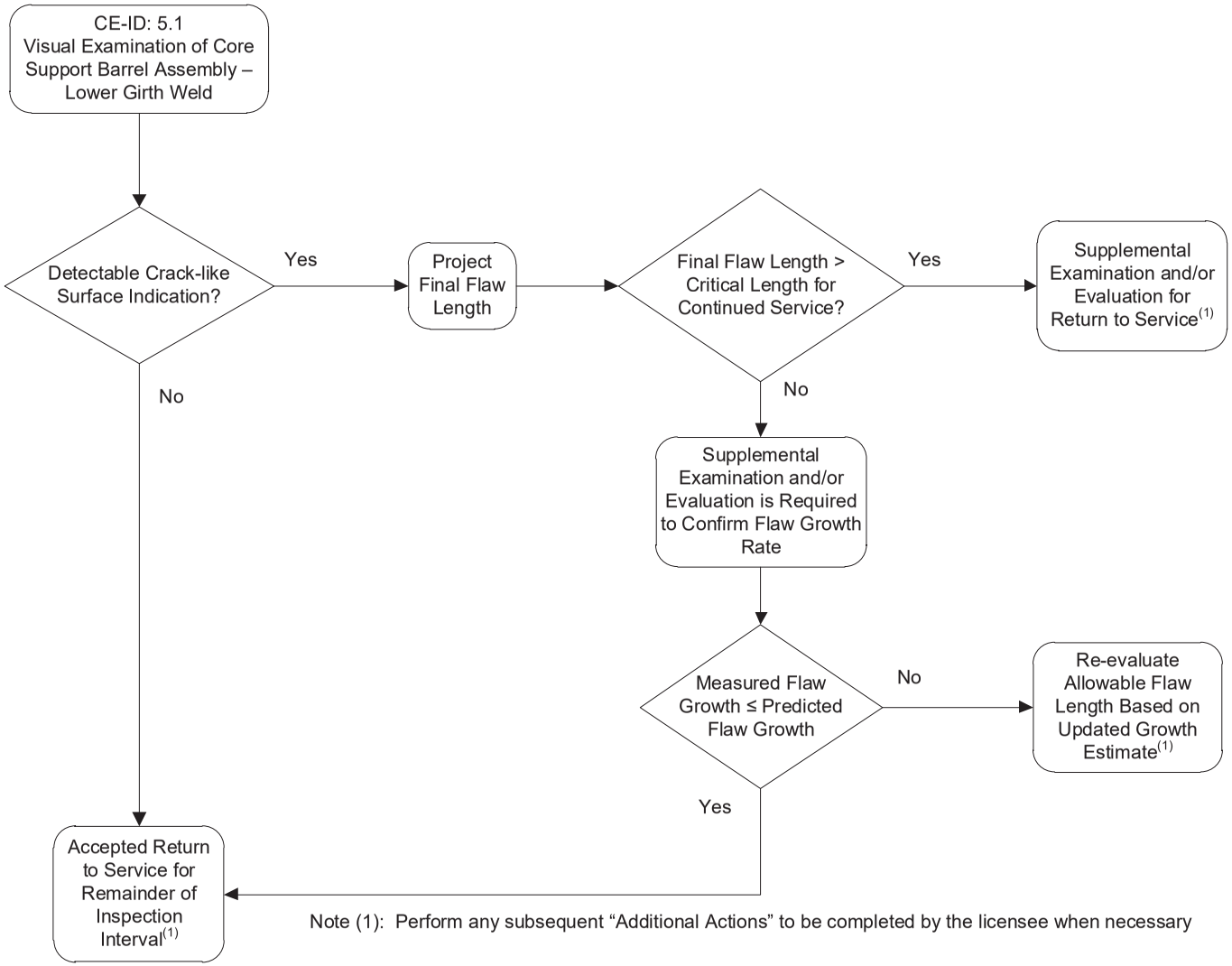
CE-ID: 5.1
Core Support Barrel Assembly – Lower Girth Weld

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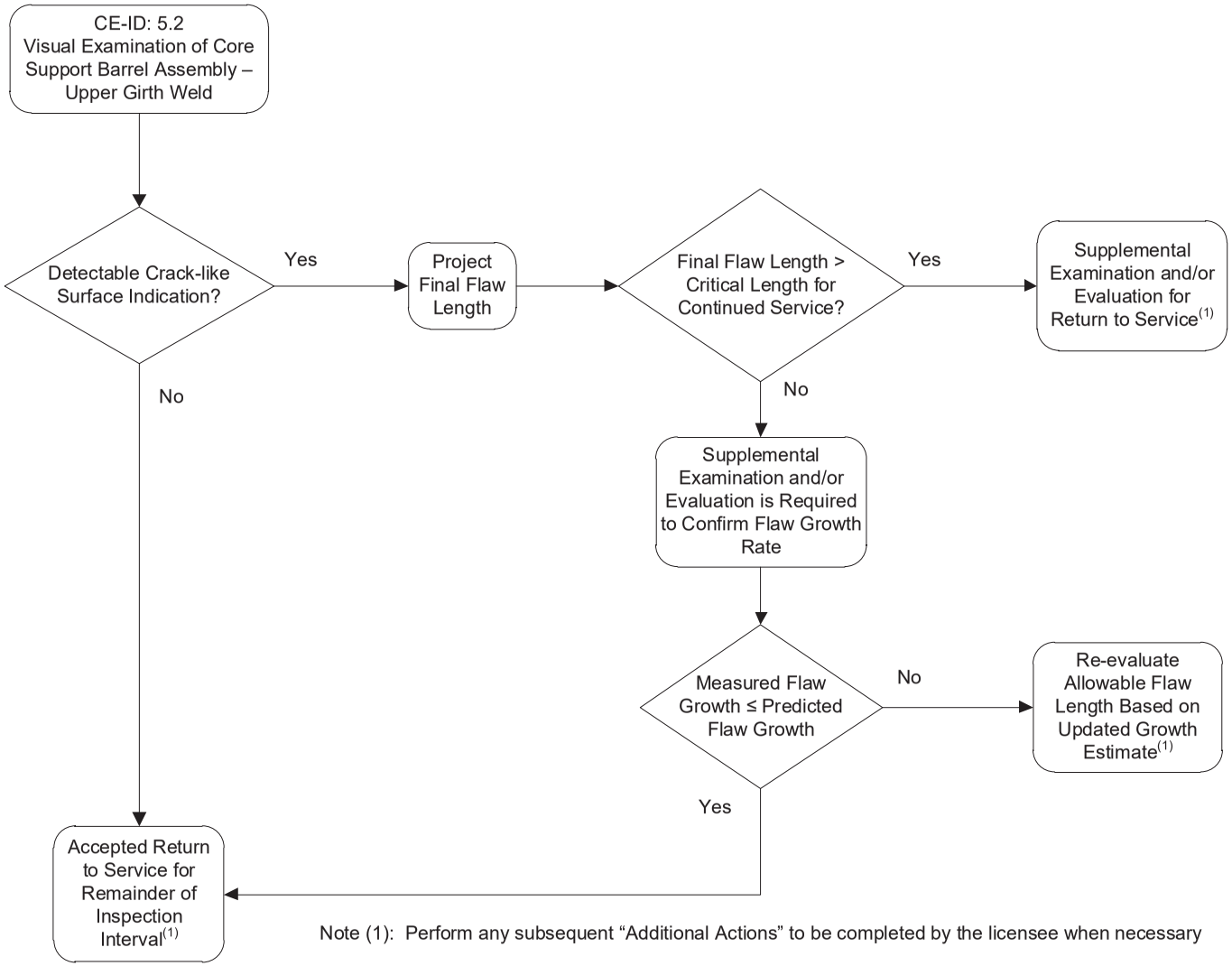
CE-ID: 5.2
Core Support Barrel Assembly – Upper Girth Weld

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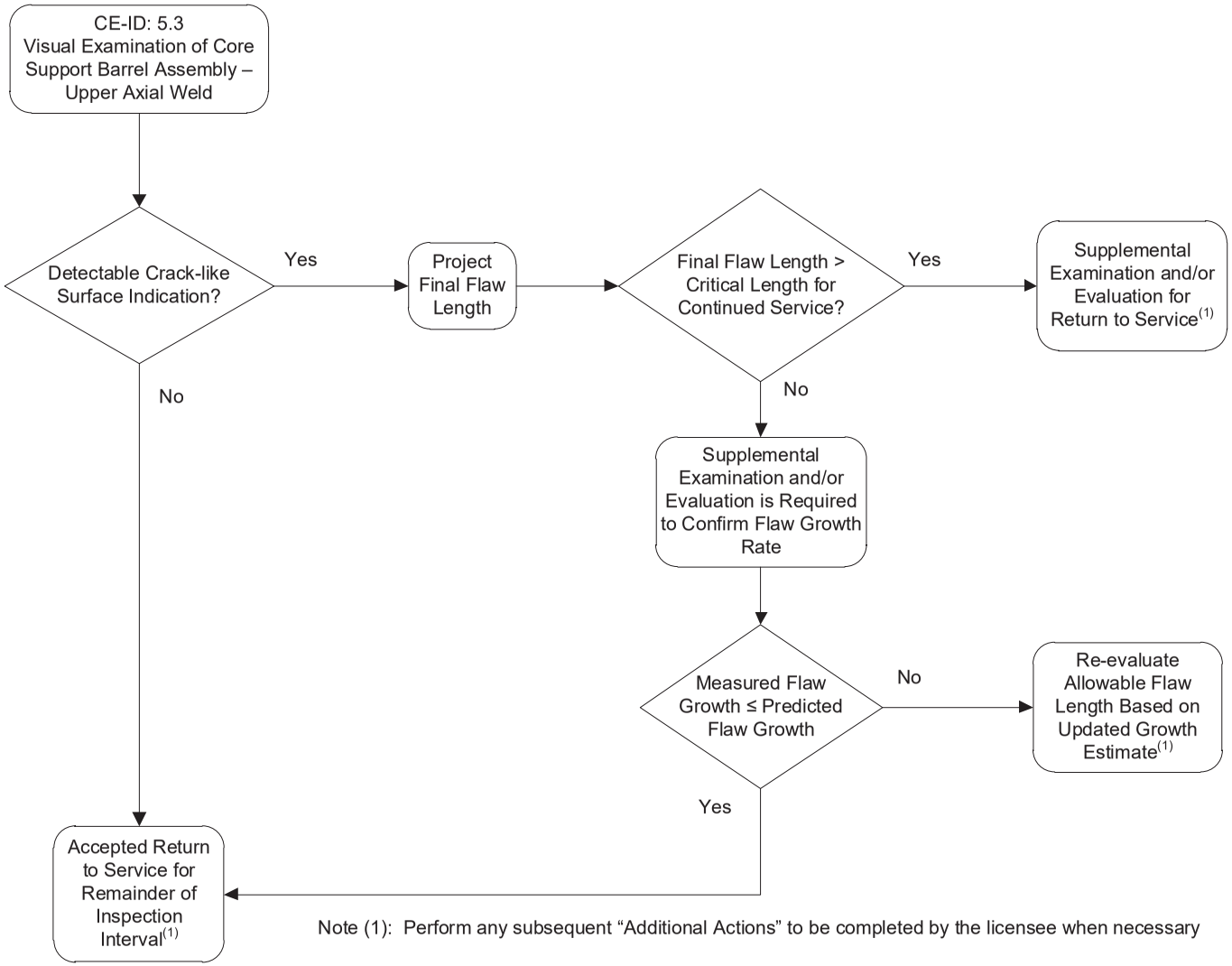
CE-ID: 5.3
Core Support Barrel Assembly – Upper Axial Weld

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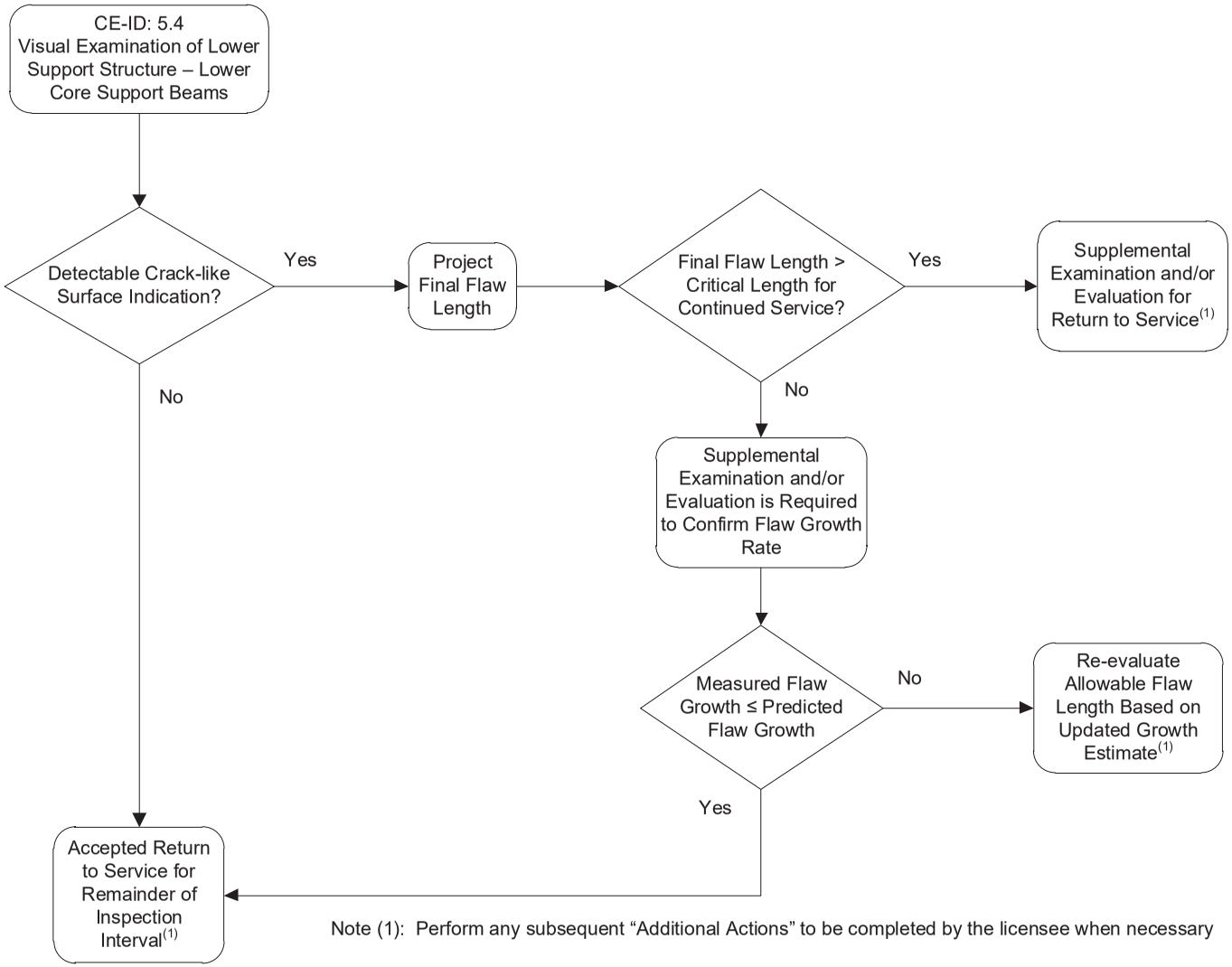
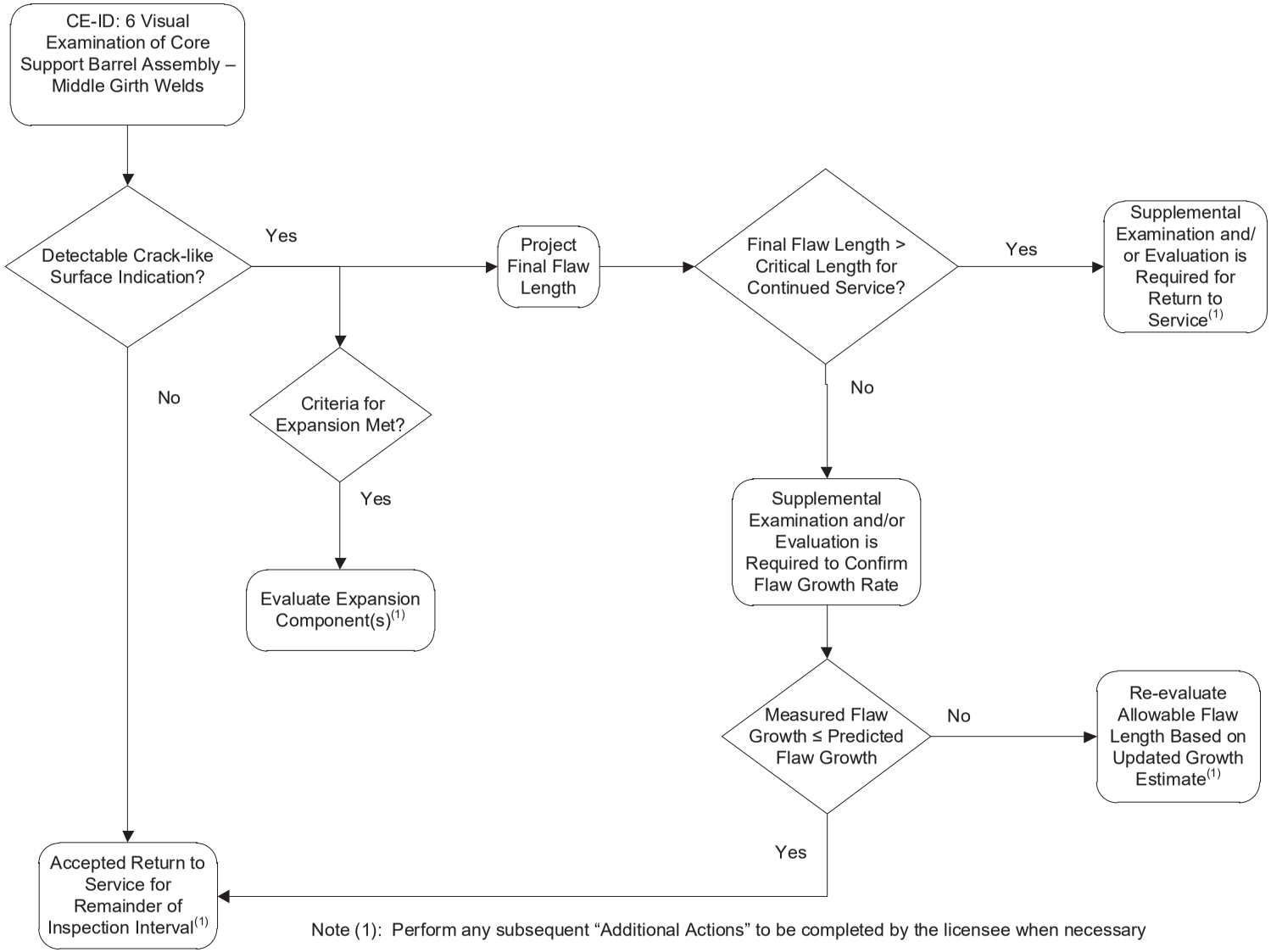
## CE-ID: 5.4
## Lower Support Structure – Lower Core Support Beams






*** This record was final approved on 08/31/2023 13:45:20. (This statement was added by the PRIME system upon its validation)

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

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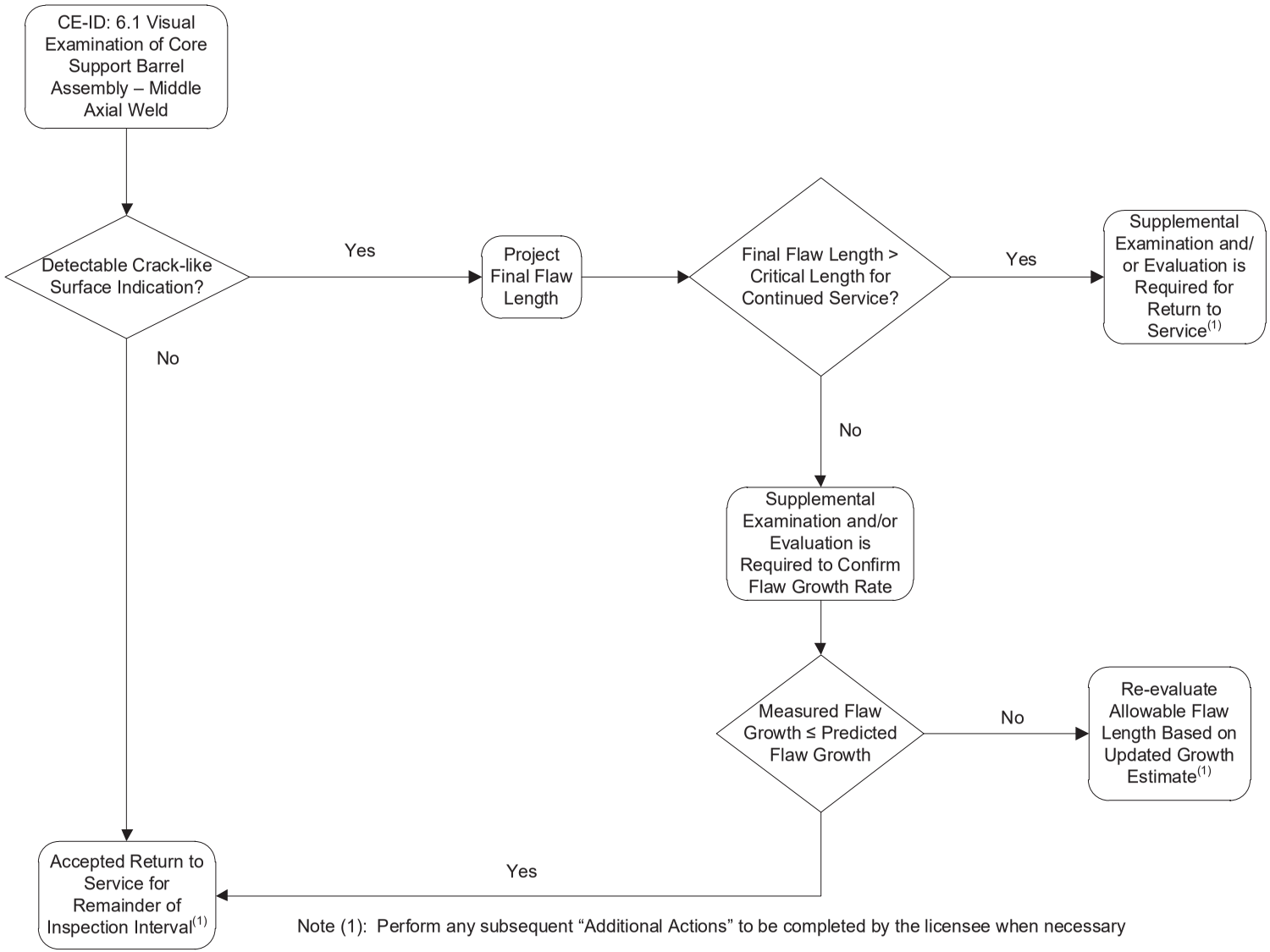
## CE-ID: 6.1
## Core Support Barrel Assembly – Middle Axial Weld

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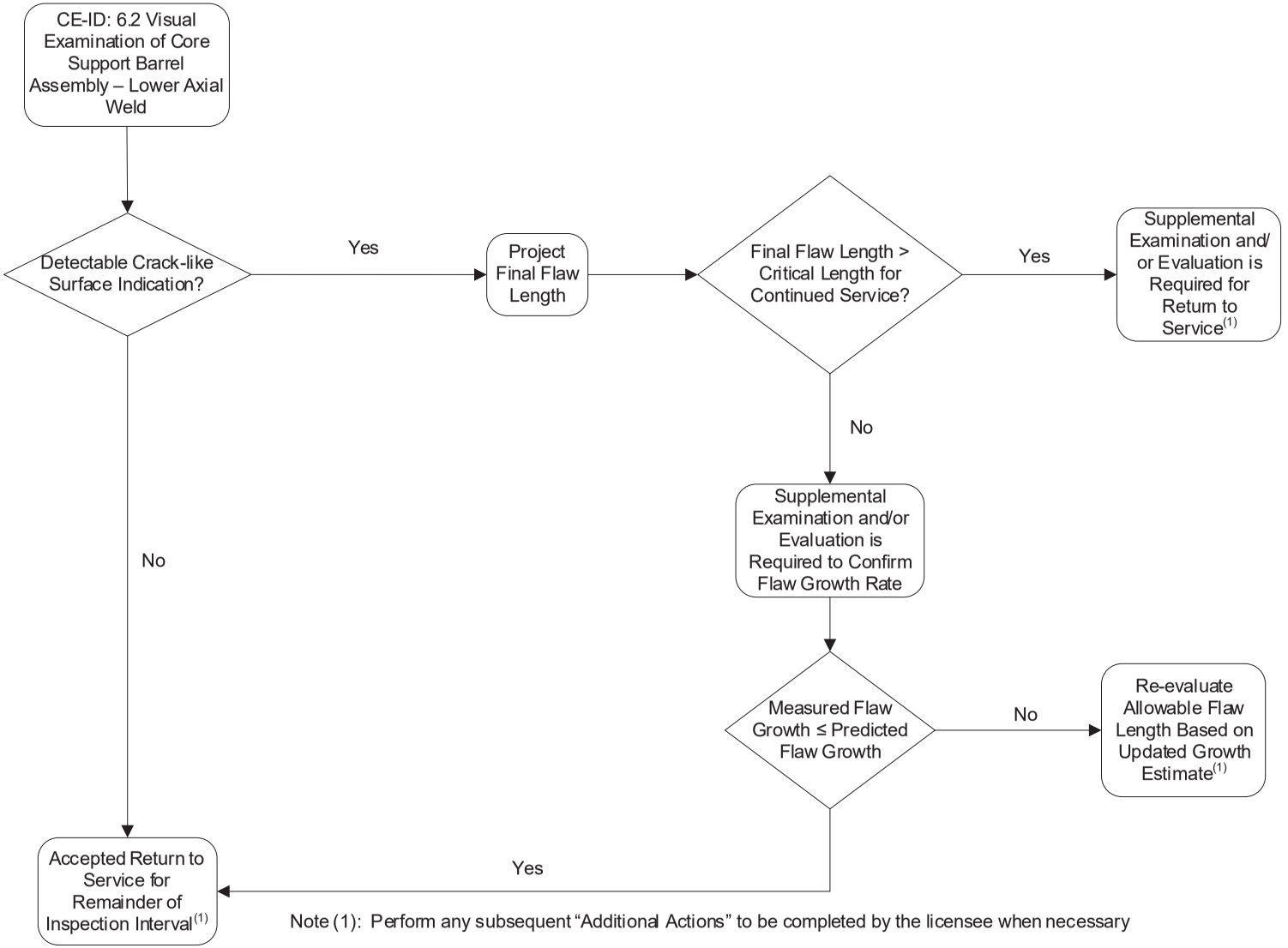
CE-ID: 6.2
Support Barrel Assembly – Lower Axial Weld

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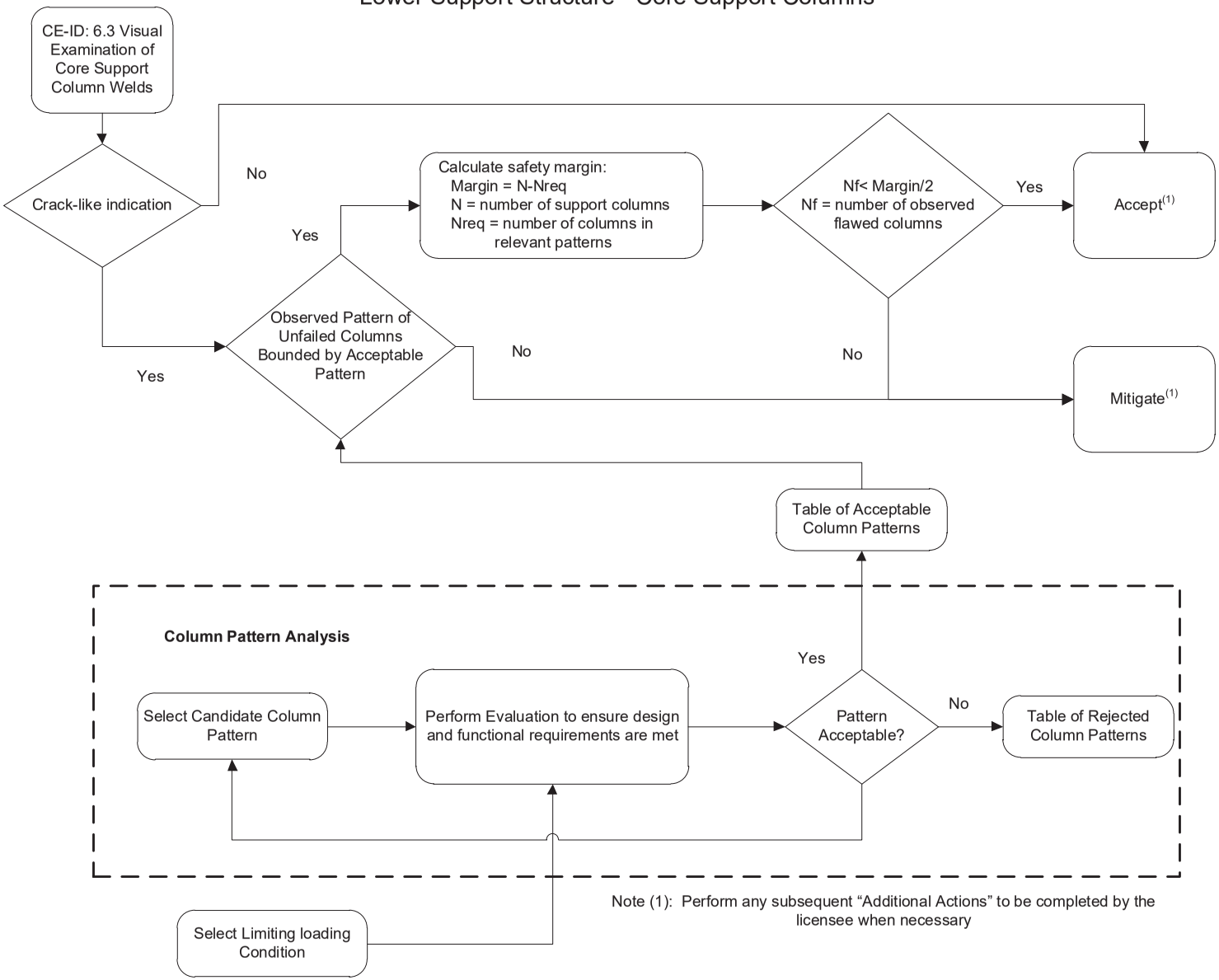
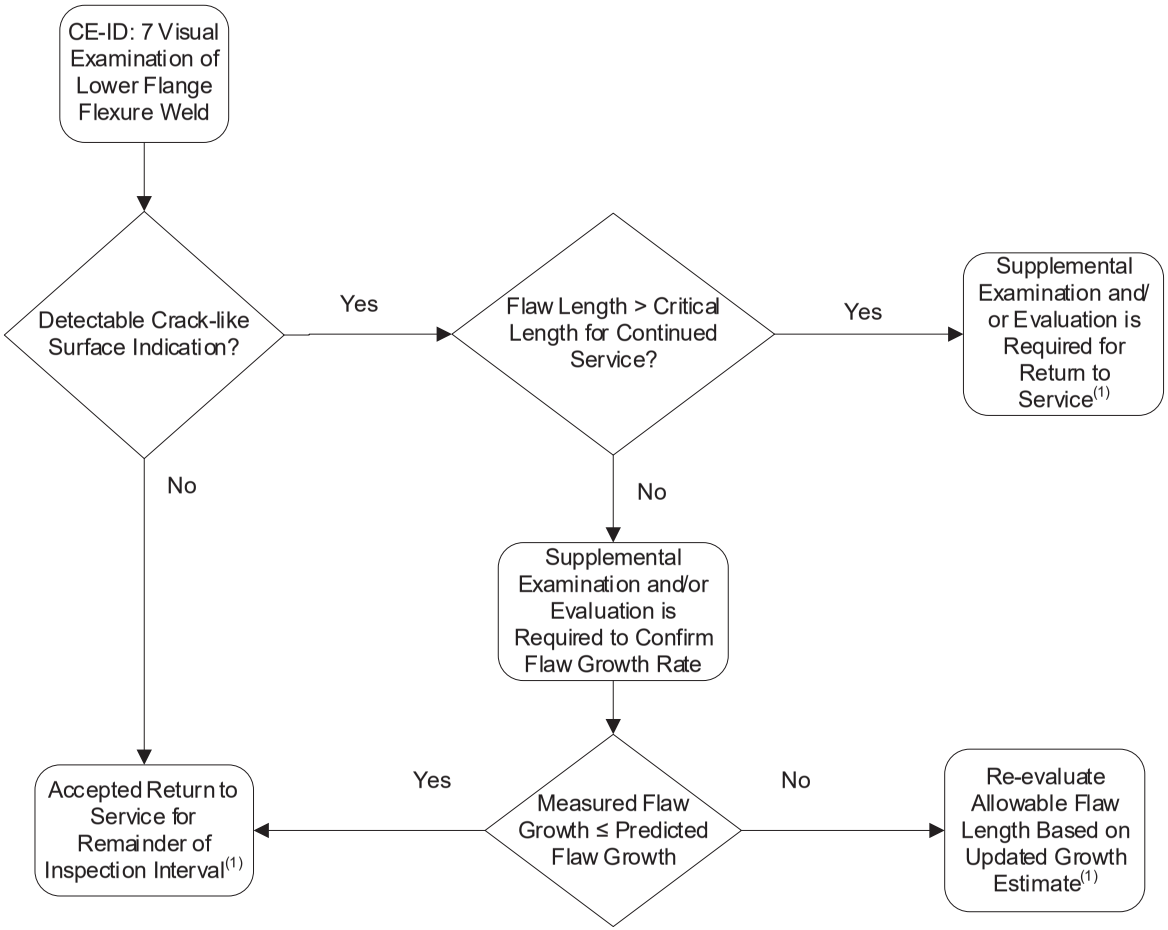
CE-ID: 6.3
Lower Support Structure - Core Support Columns






*** This record was final approved on 08/31/2023 13:45:20. (This statement was added by the PRIME system upon its validation)

# CE-ID: 7
# Core Support Barrel Assembly – CSB Flexure Weld

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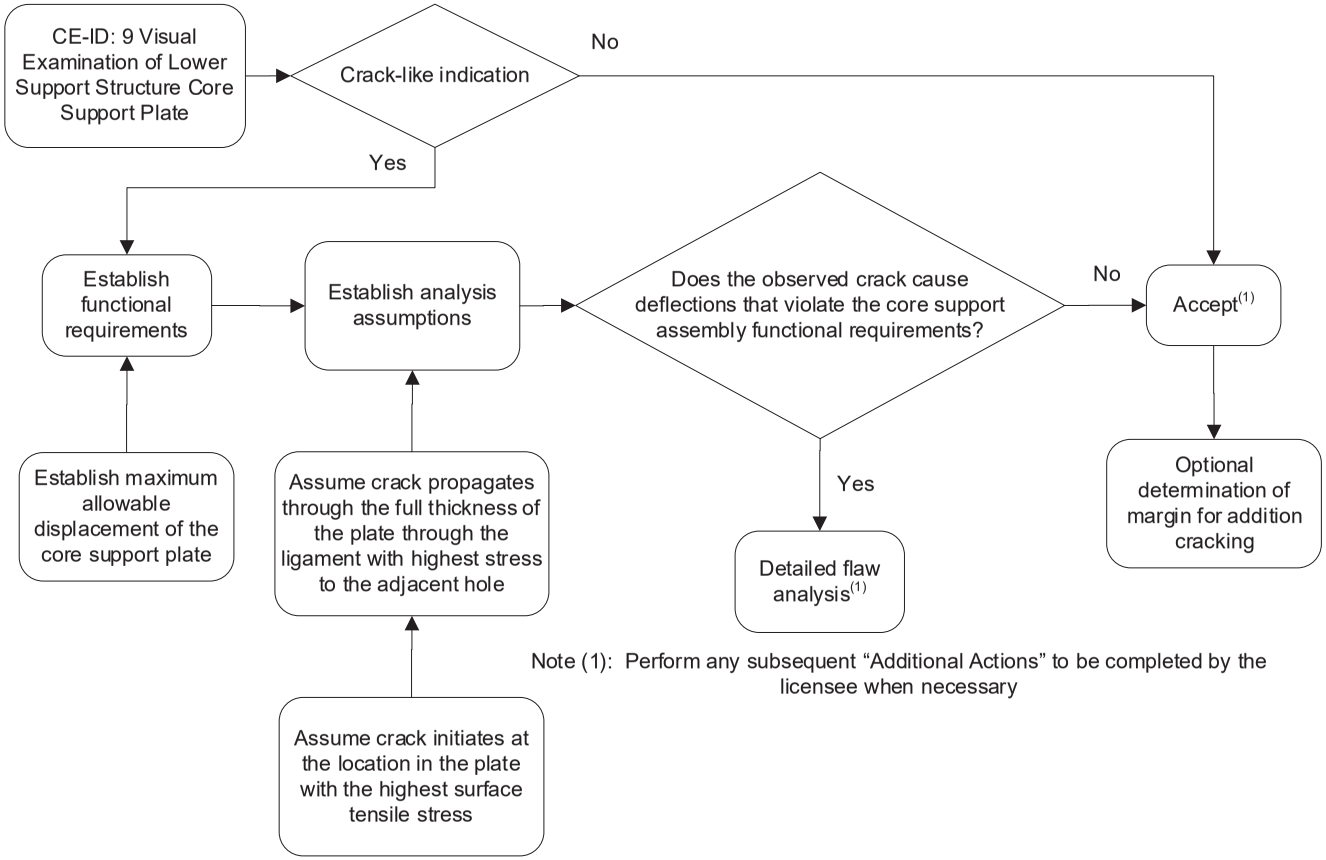
CE-ID: 9
Lower Support Structure Core Support Plate




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



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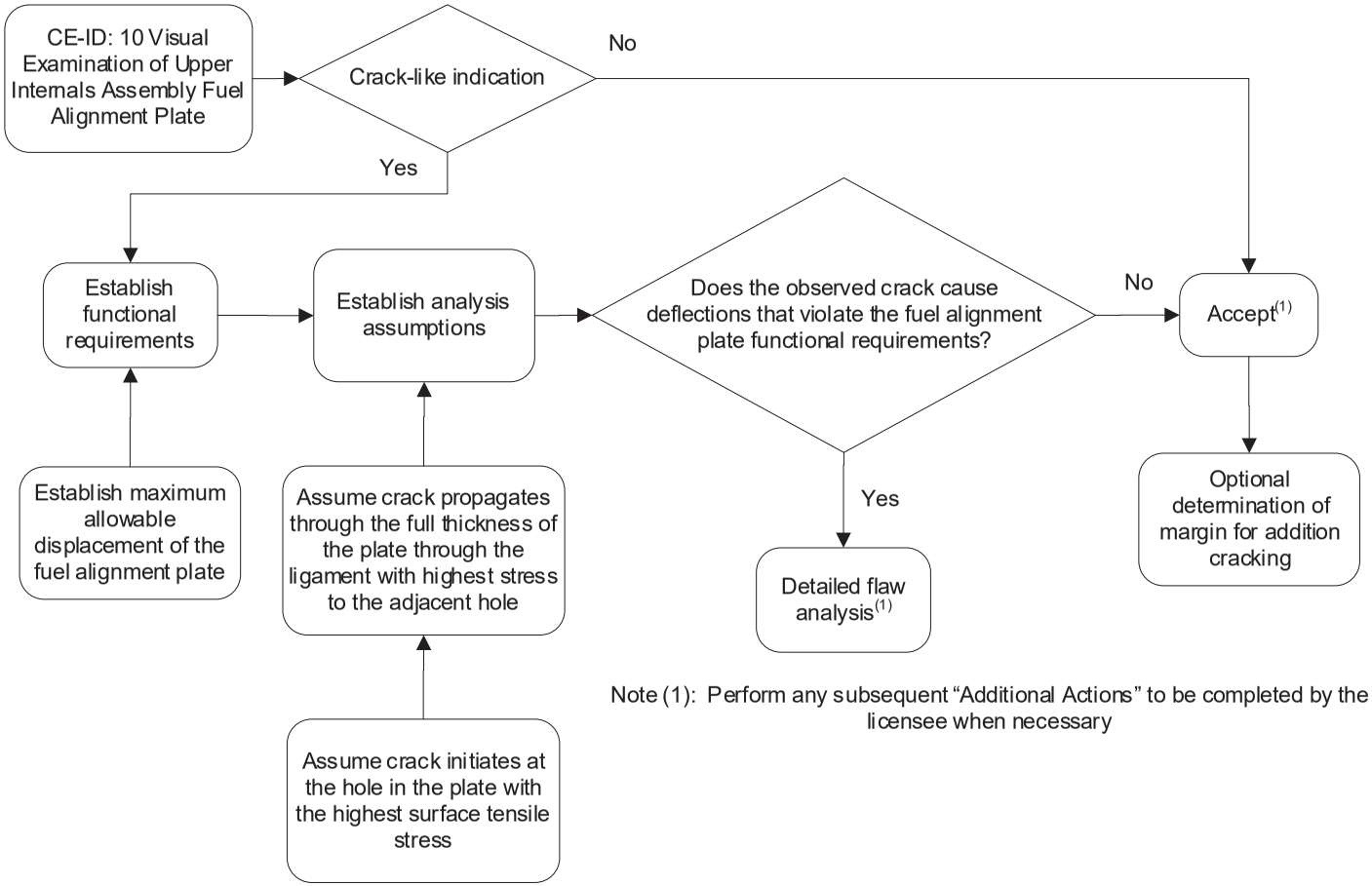
# CE-ID: 10
# Upper Internals Assembly – Fuel Alignment Plate




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



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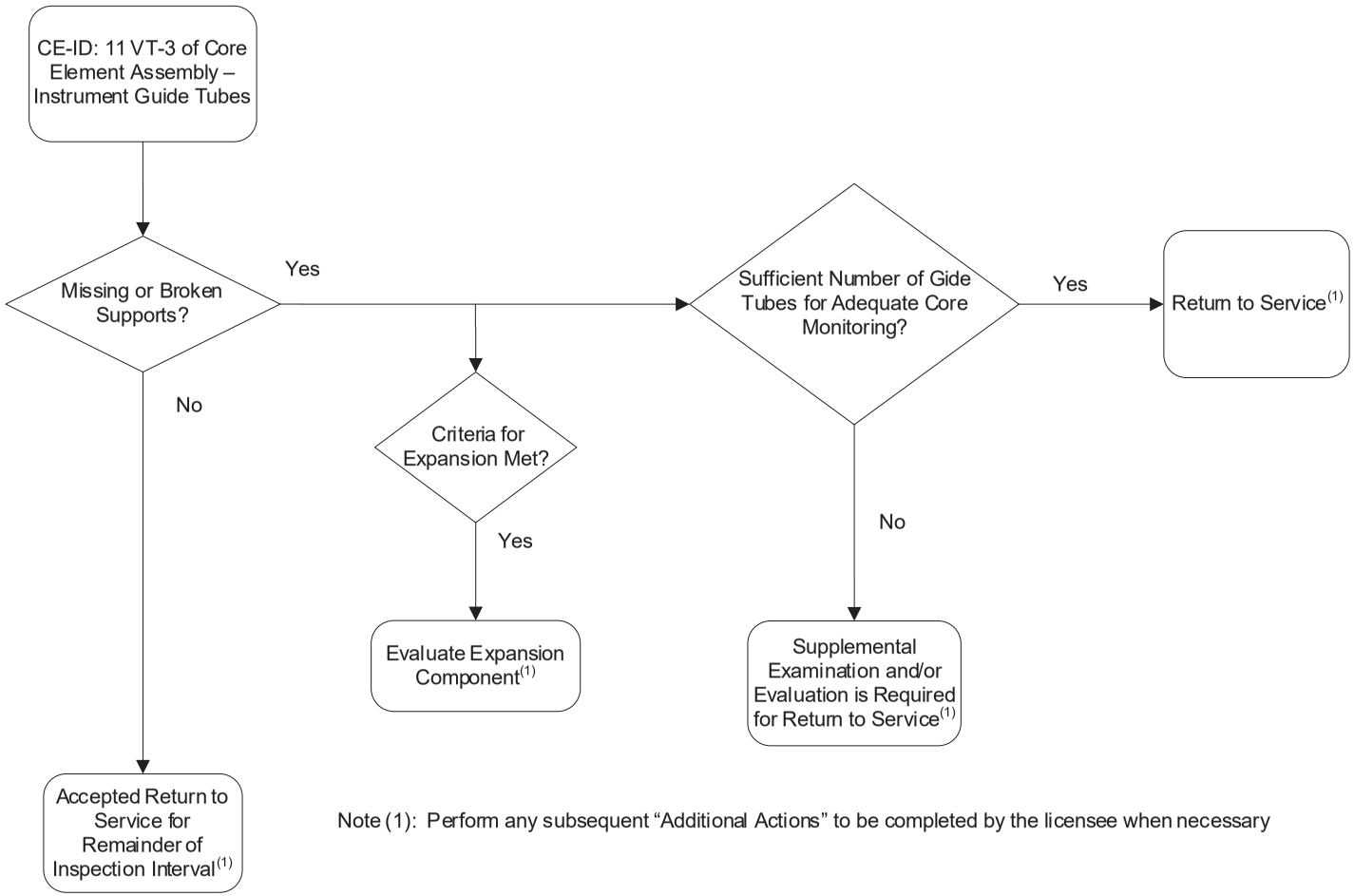
## CE-ID: 11
## Control Element Assembly – Instrument Guide Tubes




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



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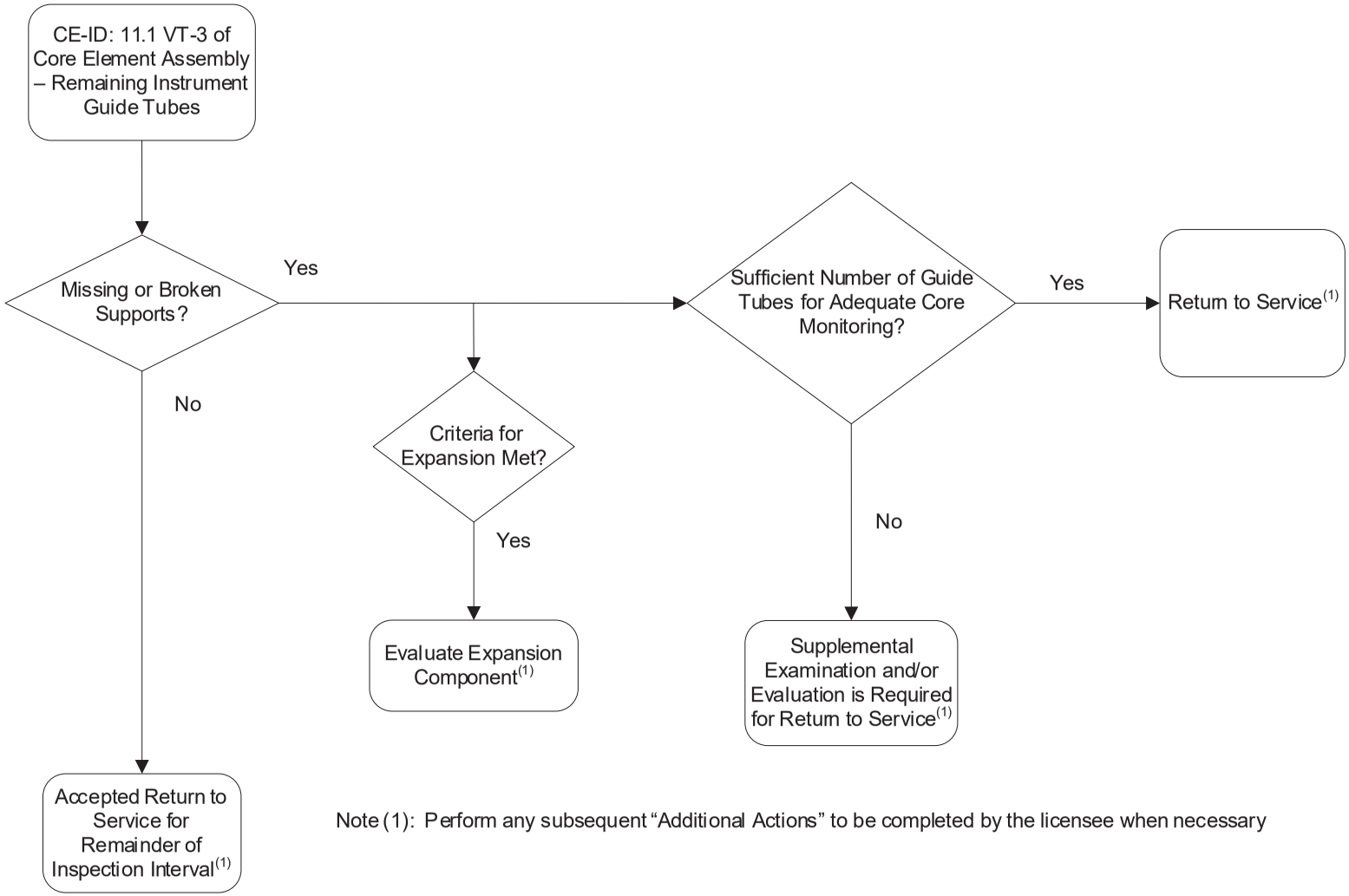
# CE-ID: 11.1
# Control Element Assembly – Remaining Instrument Guide Tubes




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



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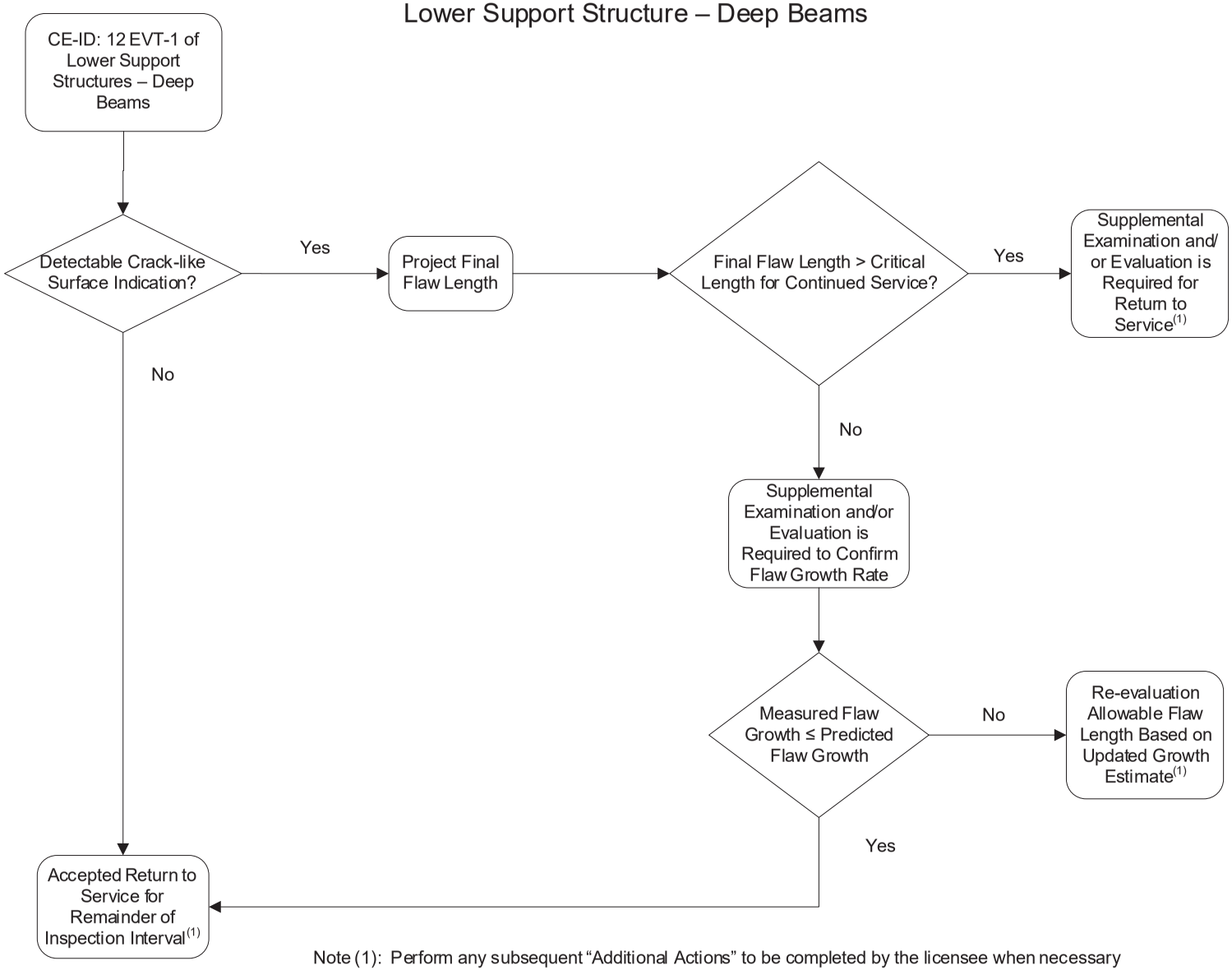
CE-ID: 12
Lower Support Structure – Deep Beams




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



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# APPENDIX E
# ACCEPTANCE CRITERIA METHODOLOGY AND DATA REQUIREMENTS FOR WESTINGHOUSE PRIMARY AND EXPANSION COMPONENT ITEMS

**Westinghouse Primary and Expansion Components**

W-ID: 1 Control Rod Guide Tube Assembly – Guide Plates (Cards)

W-ID: 2 Control Rod Guide Tube Assembly – Lower Flange Welds

- W-ID: 2.1 Control Rod Guide Tube Assembly – Remaining CRGT Lower Flange Welds
- W-ID: 2.2 Bottom-Mounted Instrumentation (BMI) System – BMI Column Bodies

W-ID: 3 Core Barrel Assembly – Upper Flange Weld

- W-ID: 3.1 Core Barrel Assembly – Upper Girth Weld
- W-ID: 3.2 Core Barrel Assembly – Upper Axial Weld
- W-ID: 3.3 Core Barrel Assembly – Lower Flange Weld
- W-ID: 3.4 Lower Internals Assembly – Lower Support Forging or Castings

W-ID: 4 Core Barrel Assembly – Lower Girth Weld

- W-ID: 4.1 Upper Internals Assembly – Upper Core Plate
- W-ID: 4.2 Core Barrel Assembly – Middle Axial Welds
- W-ID: 4.3 Core Barrel Assembly – Lower Axial Welds
- W-ID: 4.4 Lower Support Assembly – Lower Support Columns (Both Cast and Non-Cast)

W-ID: 5 Baffle-former Assembly – Baffle-Edge Bolts

W-ID: 6 Baffle-former Assembly – Baffle-Former Bolts

- W-ID: 6.1 Core Barrel Assembly – Barrel-Former Bolts
- W-ID: 6.2 Lower Support Assembly – Lower Support Column Bolts

W-ID: 7 Baffle-Former Assembly – Assembly

W-ID: 8 Alignment and Interfacing Components – Internals Hold Down Spring

W-ID: 9 Thermal Shield Assembly – Thermal Shield Flexures



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| | |
|---|---|
| **W-ID: 1** | **Control Rod Guide Tube Assembly**<br>**Guide Plates (Cards)** |
| Category: | Primary Applicability: U.S. Domestic Plants |
| Degradation Effect: | See MRP-227 Table 4-3 |
| Expansion Link: | See MRP-227 Table 4-3 |
| Function: | The control rod guide tube assembly provides alignment and an insertion path for the control rods through the upper internals. The guide tubes house guide cards and a continuous guidance section below the guide cards to provide alignment and an insertion path for control rod assemblies and to support the control rods when withdrawn. |

**Inspection**

| | |
|---|---|
| Method: | Calibrated optical technique that is capable of measuring guide card ligament thickness and slot opening width. See Figure 1 in Data Requirements section. Recommended accuracy is ±0.010 inch, however less accuracy is allowed if associated reduction in projected continued operation is recognized as discussed in Section 4.3 Item 3 of WCAP-17451-P, Rev. 2 [47].<br>Each utility shall perform an initial "baseline" examination measurement based on the inspection schedule provided in WCAP-17451-P, Rev. 2 [47]. |
| Coverage: | See WCAP-17451-P, Rev. 2 for the recommended minimum number of guide tubes to be inspected to achieve a given confidence level for the baseline inspection. |
| Observable Effect: | The observable effect of wear is the shortening of the guide card and continuous section sheath or C-tube ligaments at the four innermost rodlet holes where the shortest ligaments exist. After complete wear of the ligaments has occurred, continued wear will cause the slot width at the end of the ligaments to widen until they become at least 100 percent of the rodlet diameter and allow the rodlet to become unsupported by the guide card or some partial length of the continuous section member. |

**Failure**

| | |
|---|---|
| Failure Mechanism: | The guidance holes in the guide cards are distorted by wear (loss of material). The largest amounts of wear are typically observed in lowest guide card levels. |



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

**Guide Plates (Cards)**

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

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

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

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

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

**Methodology**

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

Data Requirements:

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



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

**Guide Plates (Cards)**

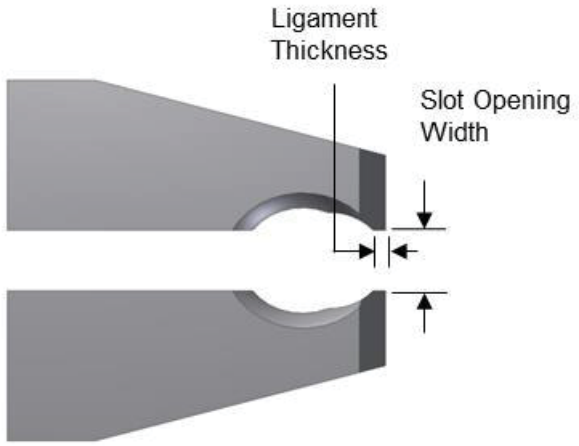


**Figure 1 Guide Card Wear Measurements**

Analysis: Operational time extension curves to project guide card wear are provided in WCAP-17451-P, Rev. 2. These curves are designed to allow a user to determine the remaining number of EFPYs (or other time unit) that a guide card can operate until the targeted amount of wear-through occurs. Procedures for operational time extension are also provided in WCAP-17451-P, Rev. 2 for the continuous guidance section. Since wear-through of one or more guide cards is allowed under certain circumstances, procedures are provided in WCAP-17451-P, Rev. 2 to determine operational time extension after wear-through occurs until the allowable number of worn-through guide cards is reached.

Re-inspection intervals are based on the observed extent of wear and the time projected to reach either the Yellow Zone or Red Zone defined in the Acceptance Criteria section. A methodology is also included in WCAP-17451-P, Rev. 2 to account for change in RCCAs i.e., from 304SS/chrome plated to ion nitride and vice versa, in the wear projection.



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

**Guide Plates (Cards)**

Acceptance Criteria:

The guide card wear criteria are defined below in Table 1. The criteria is defined in three zones—Green Zone, Yellow Zone, and Red Zone. These zone definitions should be used to both determine the applicable zone for the guide tube at the time of inspection and subsequently project to the Yellow Zone, if applicable, to plan for future wear mitigation strategy. Operation of the guide tube in the Red Zone shall be avoided. Wear mitigation contingency implementation should be planned in the Yellow Zone. The Yellow Zone range is thus defined to allow for two to three cycles to plan for appropriate contingency implementation to mitigate guide card wear.

The parameters applicable to the guide card wear criteria are defined below:

W1 = the nominal unworn guide card slot width

DR = the RCCA rodlet diameter

W4 = DR adjusted for rodlet tube wear

W3 = 0.85 x W4

n = number of allowable worn-through open guide cards for a particular guide card configuration, listed in Table 5-10 of WCAP-17451-P, Rev. 2. “Worn-through” is defined as the condition when the slot width is greater than or equal to W4.

The values for W1, W3, and DR are provided in Table 5-9 of WCAP-17451-P, Rev. 2 for the different guide tube/RCCA designs. As noted in the table, the rodlet diameter is reduced assuming a 50 percent reduction in rodlet tubing thickness due to wear.

**Table 1 Guide Card Wear Criteria Zones**

| Criterion Zone | Range | |
|---|---|---|
| | Start | End |
| Red Zone | End of Red Zone minus two EFPY | Slot width equal to or greater than $W_3$ for (n+1) consecutive guide cards |
| Yellow Zone | Start of Red Zone minus five EFPY | Start of Red Zone |
| Green Zone | No wear | Start of Yellow Zone |



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## W-ID: 1 Control Rod Guide Tube Assembly Guide Plates (Cards)

Alternative Wear Criteria:

An alternate wear criteria that is easier to implement as compared to the one defined in Table 1 is provided for use. This alternative criteria is however more conservative than the wear criteria defined in Table 1. The criteria zones are defined in Table 2.

The required and recommended actions for each guide card wear criterion zone are defined in Table 3.

**Table 2 Alternative Guide Card Wear Criteria Zones**

| Criterion Zone | Range | |
|---|---|---|
| | Start | End |
| Red Zone | End of Red Zone minus two EFPY | Slot width equal to $W_3$ at any one card |
| Yellow Zone | Start of Red Zone minus five EFPY | Start of Red Zone |
| Green Zone | No wear | Start of Yellow Zone |

**Table 3 Guide Card Wear Criteria Zones and Actions**

| Criterion Zone | Recommended or Required Actions |
|---|---|
| Green Zone | Recommendations:<br>For a guide tube determined to be in this zone, the next maintenance inspection period will be determined from predictive wear calculations based on the baseline data. The predictive wear calculation will set the next inspection period in the Yellow Zone. |
| Yellow Zone | Recommendations:<br>For a guide tube determined to be in this zone, consider the following options:<br>• Replace or exchange the guide tube with a part-length or spare guide tube<br>• Performing engineering evaluation to determine continued operation<br>• Determining subsequent inspection periods to preclude operation in the Red Zone<br>Refer to [27] for a detailed list of contingency options that could be implemented to avoid operating in the Red Zone. |
| Red Zone | Required:<br>For a guide tube determined to be in this zone, rodded use of the guide tube without modifications shall be discontinued to preclude wear-through at (n+1) guide cards. |

Table 4 contains lower continuous guidance member criterion zones and recommended or required actions. Acceptance criteria are for wear at the top end of the lower continuous guidance section members where wear is most critical for guidance above the continuous section and where rodlet jamming is most likely to occur. A relative view of the wear along the lower continuous guidance section members' length – where ligament depth variations can be viewed – can be video recorded. Parameters for evaluating the wear criterion zones in 17x17 standard, AS, and XL guide tubes are as follows:



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

**Guide Plates (Cards)**

W1 = nominal unworn continuous guidance slot width

W3 = slot width = 0.85 x W4

W4 = slot width = DR adjusted for rodlet tube wear

DR = RCCA rodlet diameter

Similar criteria for the 14x14 and 15x15 continuous guidance section members are likely possible, but have not been firmly established. Therefore, for the 14x14 and 15x15 guide tubes, the following more conservative criteria are recommended. If it becomes apparent that more relaxed criteria for the continuous guidance members are needed, a plant-specific evaluation would be required if generic development of these criteria has not been established at a later time.

Therefore, in Table 4 for 14x14 and 15x15 guide tubes:

W1 = zero wear depth

W3 = ligament wear depth =85 percent of ligament thickness

W4 = ligament wear depth = 100 percent of ligament thickness



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

## Guide Plates (Cards)

**Table 4 Continuous Guidance Section Wear Criteria Zones and Actions**

| Criteria Zone | Zone Range | | Recommended or Required Actions |
|---|---|---|---|
| | Start | End | |
| Red Zone | End of Red Zone minus two EFPY | Ligament or slot width equal to $W_4$ | Required:<br>For any measured ligament wear in this zone, rodded use of the guide tube without modifications shall be discontinued to prevent the slot width from opening to the point where rodlet break-through can occur along part of the length of the continuous guidance member. Rodlet gouging and jamming in the continuous guidance slot could result under such a wear condition. |
| Yellow Zone | Start of Red Zone minus five EFPY | Start of Red Zone | Recommendations:<br>For any measured ligament wear in this zone, consider options such as:<br>• Replace or exchange the guide tube with a part-length or spare guide tube<br>• Performing engineering evaluation to determine continued operation<br>• Determining subsequent inspection periods to preclude operation in the Red Zone<br>Perform rod drop time trending(1) relative to rod drop technical specification limits. |
| Green Zone | No wear | Start of Yellow Zone | Recommendations:<br>For any measured ligament wear in this zone, determine the next maintenance inspection period from predictive wear calculations based on the baseline data. The predictive wear calculation will set the next inspection period in the Yellow Zone. Review rod drop times for possible trending relative to technical specification limits that may preclude the wear limits. |

**Note:**

1. It is postulated that high wear in the continuous section could have an impact on rod drop time for the particular guide tube. Therefore, the recommendation here is to review the rod times for these guide tubes to determine if a significant change in rod drop time is noticed or not. This recommendation does not require performing any rod drop tests in addition to what a plant already performs prior to plant start-up.

Approach: Generic work completed in existing PWROG program WCAP-17451-P Rev. 2.

Additional Action(s): None



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| | |
|---|---|
| **W-ID: 2** | **Control Rod Guide Tube Assembly**<br>**Lower Flange Welds** |
| Category: | Primary Applicability: All plants |
| Degradation Effect: | See MRP-227 Table 4-3 |
| Expansion Link: | See MRP-227 Table 4-3 |
| Function: | The control rod guide tube assembly provides alignment and insertion path for control rods through upper internals. The lower flange welds retain the structural alignment of the component. Guide tubes must maintain rod stability in normal and LOCA transients. |
| **Inspection** | |
| Method: | Visual examination. See MRP-227 Table 4-3. |
| Coverage: | See MRP-227 Table 4-3 |
| Observable Effect: | Any individual weld with observed crack must be assumed to have failed. The vertical beam portion appears to be out of position. |
| **Failure** | |
| Failure Mechanism: | Flow in the upper head applies bending moment to control rod guide tube assembly. Maximum bending stresses tend to occur near top of continuous guidance section. Stresses may lead to formation of SCC or fatigue cracks. Weld cracking may lead to loss of stiffness in guide tube assembly and loss of support capability. |
| Failure Effect: | Loss of structural stability. Excessive deflection could impede control rod assembly insertion. |
| Failure Criteria: | Design limits on the CRGT assembly are generally expressed as a maximum allowable load, which is determined based on the assembly compliance. This analysis implies a maximum allowable deflection. Interference between the guide cards and the guide tubes occur when the deflection exceeds this limit. |
| **Methodology** | |
| Goal: | The goal of the acceptance criteria is to evaluate the adequacy of any CRGT found to have failed lower flange welds. The stiffness of a CRGT assembly with failed welds must be sufficient to maintain allowable deflections when LOCA and SSE loads are applied. Allowable load on control rod guide tube assembly is defined by empirical testing. |
| Data Requirements: | Loads<br>Finite element model of lower CRGT assembly to evaluate weld failures calibrated to benchmark data |
| Analysis: | 1. Determine design basis assumptions for CRGT assembly (maximum allowable load, assembly compliance).<br>2. Lower section must be modeled in detail; upper sections may be treated as large beams. |



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**W-ID: 2** **Control Rod Guide Tube Assembly**

**Lower Flange Welds**

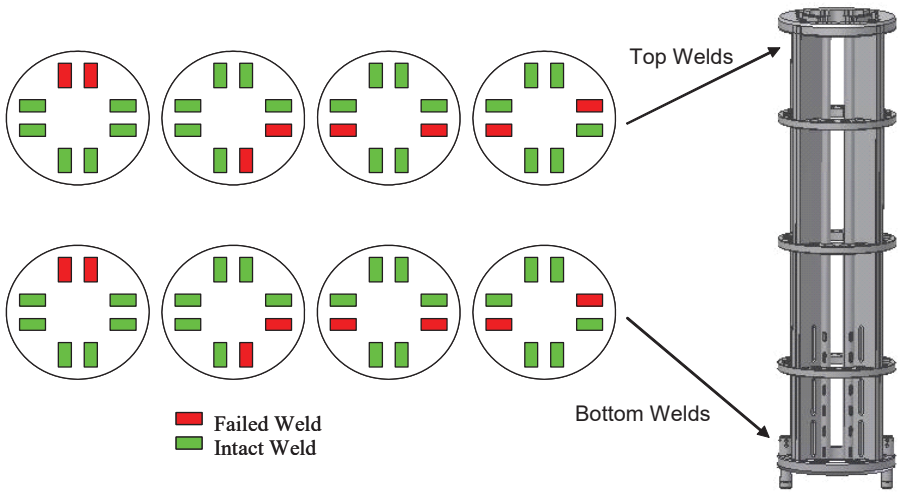
3. Calibrate FEA model and boundary constraints against design basis assumptions.
4. Remove test pattern of welds.
5. Run FEA.
6. If deflection is greater than limit in Step 1, pattern is not acceptable.
7. Iterate Steps 4–6 to create library of acceptable and unacceptable patterns.
8. Match patterns to field observations assuming that any weld with flaw has failed.
9. Should be able to observe sufficient number of welds to demonstrate that assembly is acceptable.

Acceptance Criteria:

This acceptance criteria is based on a minimum number of welds that must continue to function (without cracking) to allow scramming of the control rods in the event of combined LOCA and SSE.

The inspection plan for the CRGTs is a sampling approach. If indications are detected then evaluation of the CRGT(s) with the indication must be performed. This may also lead to an increase of the sample being inspected, or subsequent additional inspections.

This sampling approach is cognizant of industry operating experience and if no indications are found in the inspected reactor then there is no assumption that uninspected welds in that reactor may have flaws.




Approach:

Plant-specific analysis due to large variety of sizes and designs. There may be some potential for smaller plant groupings.

Additional Action(s):

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



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| W-ID: 2.1 | Control Rod Guide Tube Assembly<br>Remaining Lower Flange Welds |
|---|---|
| Category: | Expansion Applicability: All plants |
| Degradation Effect: | See MRP-227 Table 4-6 |
| Primary Link: | See MRP-227 Table 4-6 |
| Function: | The control rod guide tube assembly provides alignment and insertion path for control rods through upper internals. The lower flange welds retain the structural alignment of the component. Guide tubes must maintain rod stability in normal and LOCA transients. |
| **Inspection** | |
| Method: | Visual examination. See MRP-227 Table 4-6. |
| Coverage: | See MRP-227 Table 4-6 |
| Observable Effect: | Any individual weld with observed crack must be assumed to have failed. The vertical beam portion appears to be out of position. |
| **Failure** | |
| Failure Mechanism: | Flow in the upper head applies bending moment to control rod guide tube assembly. Maximum bending stresses tend to occur near top of continuous guidance section. Stresses may lead to formation of SCC or fatigue cracks. Weld cracking may lead to loss of stiffness in guide tube assembly and loss of support capability. |
| Failure Effect: | Loss of structural stability. Excessive deflection could impede control rod assembly insertion. |
| Failure Criteria: | Design limits on the CRGT assembly are generally expressed as a maximum allowable load, which is determined based on the assembly compliance. This analysis implies a maximum allowable deflection. Interference between the guide cards and the guide tubes occur when the deflection exceeds this limit. |
| **Methodology** | |
| Goal: | The goal of the acceptance criteria is to evaluate the adequacy of any CRGT found to have failed lower flange welds. The stiffness of a CRGT assembly with failed welds must be sufficient to maintain allowable deflections when LOCA and SSE loads are applied. Allowable load on control rod guide tube assembly is defined by empirical testing. |
| Data Requirements: | Loads<br>Finite element model of lower CRGT assembly to evaluate weld failures calibrated to benchmark data |
| Analysis: | 1. Determine design basis assumptions for CRGT assembly (maximum allowable load, assembly compliance).<br>2. Lower section must be modeled in detail; upper sections may be treated as large beams. |



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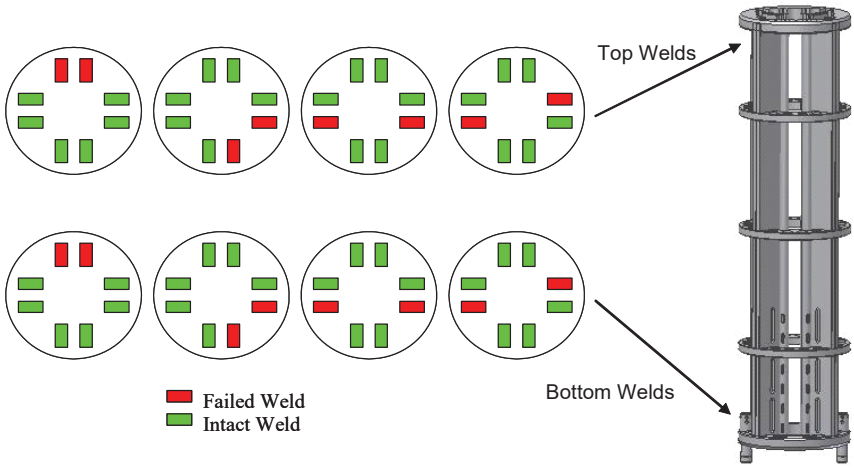
3. Calibrate FEA model and boundary constraints against design basis assumptions.
4. Remove test pattern of welds.
5. Run FEA.
6. If deflection is greater than limit in Step 1, pattern is not acceptable.
7. Iterate Steps 4–6 to create library of acceptable and unacceptable patterns.
8. Match patterns to field observations assuming that any weld with flaw has failed.
9. Should be able to observe sufficient number of welds to demonstrate that assembly is acceptable.

Acceptance Criteria:

This acceptance criteria is based on a minimum number of welds that must continue to function (without cracking) to allow scramming of the control rods in the event of combined LOCA and SSE.

The inspection plan for the CRGTs is a sampling approach. If indications are detected then evaluation of the CRGT(s) with the indication must be performed. This may also lead to an increase of the sample being inspected, or subsequent additional inspections.

This sampling approach is cognizant of industry operating experience and if no indications are found in the inspected reactor then there is no assumption that uninspected welds in that reactor may have flaws.




Approach:

Plant-specific analysis due to large variety of sizes and designs. There may be some potential for smaller plant groupings.

Additional Action(s):

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



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| | |
|---|---|
| **W-ID: 2.2** | **Bottom Mounted Instrumentation System**<br>**Bottom-mounted Instrumentation (BMI) Column Bodies** |
| Category: | Expansion Applicability: All plants |
| Degradation Effect: | See MRP-227 Table 4-6 |
| Primary Link: | See MRP-227 Table 4-6 |
| Function: | The BMI columns define the path for flux thimbles to be inserted into the fuel assemblies.<br>Flux thimbles are normally withdrawn prior to refueling and re-inserted at the end of refueling.<br>The plant must maintain a required number of functioning flux thimbles for core mapping. |
| **Inspection** | |
| Method: | See MRP-227 Table 4-6 |
| Coverage: | See MRP-227 Table 4-6 |
| Observable Effect: | Fracture should be readily visible.<br>Large loose parts.<br>Skewed flow.<br>Weakened support. |
| **Failure** | |
| Failure Mechanism: | The BMI columns may be subject to fatigue due to either thermal fatigue or flow-induced vibrations. |
| Failure Effect: | Inability to insert flux thimbles. This effect would be noted during refueling outage. Consequences of failure during ensuing operating period are believed to be minimal. |
| Failure Criteria: | The plant must maintain a required number of functioning flux thimbles for core mapping.<br>Any BMI column with an observable crack will be assumed to have failed.<br>The primary pressure boundary must be intact. |
| **Methodology** | |
| Goal: | Configuration of unfailed BMI columns should be sufficient to allow required flux mapping. |
| Data Requirements: | Criteria should be part of the plant-specific technical specifications for plant operation. |



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| | |
|---|---|
| **W-ID: 2.2** | **Bottom Mounted Instrumentation System**<br>**Bottom-mounted Instrumentation (BMI) Column Bodies** |
| Analysis: | Evaluate stability of failed BMI column. Any section that could potentially detach and become a loose part or otherwise interfere with plant operation should be removed or stabilized. |
| Acceptance Criteria: | Plant must have minimum number of unfailed BMI assemblies to allow flux-mapping at startup. |
| Approach: | Pass/Fail inspection with established minimum number of instrumentation tubes. Based directly on plant technical specifications. |
| Additional Action(s): | None |



*** This record was final approved on 08/31/2023 13:45:20. (This statement was added by the PRIME system upon its validation)

| W-ID: 3 | Core Barrel Assembly<br>Upper Flange Weld | | |
|---|---|---|---|
| Category: | Primary | Applicability: | All plants |
| Degradation Effect: | See MRP-227 Table 4-3 | | |
| Expansion Link: | See MRP-227 Table 4-3 | | |
| Function: | Primary core support structure | | |

**Inspection**

| | |
|---|---|
| Method: | See MRP-227 Table 4-3 |
| Coverage: | See MRP-227 Table 4-3 |
| Observable Effect: | The specific relevant condition is a detectable crack-like surface indication. |

**Inputs and Assumptions**

There are several inputs and assumptions that are critical to the development of acceptance criteria for the RV internals upper flange weld. These items are stated below:

- The inspections identified in MRP-227 are intended to provide a sampling of potential locations of degradation. Under this approach, inspection of one side (surface) of the weld is assumed to provide an adequate sampling for monitoring of SCC.
- The change in resistance to fracture of the RV upper flange weld can be correlated to the accumulated fluence at each weld location. Welds that are subject to low fluence are considered to have a high degree of resistance to fracture. Correspondingly, those welds subject to high fluence have lower resistance to fracture.
- It is assumed that the primary mode of crack growth to be expected in this region is stress corrosion cracking.
- The prediction of crack growth is based on the stress intensity factor, K, calculated using linear elastic fracture mechanics. The rate of crack growth is dependent on the amount of neutron fluence that the weld is expected to accumulate over the licensed operating lifetime. Since there has been very limited experience of SCC initiated cracks in operating PWRs to date, growth rates developed for the prediction of SCC in boiling water reactors (BWRs) are assumed to be appropriate for prediction of crack growth due to SCC in PWR reactor internals.
    - For weld locations subjected to fluence levels less than or equal to $5x10^{20}$ n/cm$^2$ (E>1MeV)(0.75 dpa), the maximum of the crack growth rate in 2017 ASME Section XI Appendix C-8520 **or** the crack growth equation in ASME Code Case N-889 [52] will be used. The conditions for the use of Code Case N-889 identified in Regulatory Guide RG 1.147, Rev. 20 [67] must be met.
    - For fluence levels above $5x10^{20}$ n/cm$^2$ (E>1MeV)(0.75 dpa) up to and including $1.4x10^{22}$ n/cm$^2$ (E>1 MeV) (20 dpa), the crack growth equation in ASME Code Case N-889 [52] will be used. The conditions for the use of Code Case N-889 identified in



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| | |
|---|---|
| **W-ID: 3** | **Core Barrel Assembly**<br>**Upper Flange Weld** |

Regulatory Guide RG 1.147, Rev. 20 [67] must be met for the development of acceptance criteria.

- Depending on the magnitude of cyclic loads, such as thermal transients, additional crack growth as a result of these loads may need to be considered
- Acceptance criteria can be developed for the entire operating license of a given plant by using predicted end-of-license fluence values. However, re-inspection on the recommended intervals is required to confirm that the acceptance criteria are met. If there are changes to these fluence projections, such as in the event of a power uprate or change in core loading pattern, it would be necessary to confirm that the inputs selected based on fluence, such as SCC growth rate and fracture toughness, remain applicable or bounding until the end of the operating license.

**Failure**

| | |
|---|---|
| Failure Mechanism: | Cracking (SCC) |
| Failure Effect: | Potential loss–of-core support. |
| Failure Criteria: | Actively growing through-wall flaws may require mitigation.<br>An existing flaw is unacceptable if the flaw length projected at the next inspection cycle allows a potential loss-of-core support. |

**Methodology**

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

Data Requirements:

1. Surface crack length, as determined by visual inspection:
   a. The 2013 edition of the ASME Code Section XI: IWA-3330(a) and Figure IWA-3330-1 provide requirements for the minimum allowable separation distance that will allow multiple cracks to be treated individually. This approach is part of the guidance in this report. The requirement in Section XI is that the ligament between adjacent cracks must be greater than half of the thickness of the material. If this criterion is not met, the individual crack lengths and the length of the ligament between the cracks must be summed. This total length is then compared to the allowable length. Note that the 2007 and 2010 editions of ASME Code Section XI have the equivalent content to the 2013 edition and are also applicable.
   b. The flaw proximity rules in BWRVIP-158 [62] may be optionally applied in lieu of those defined in ASME Section XI: IWA-3330(a) and Figure IWA-3330-1 when appropriate.
2. Flaw Depth



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**W-ID: 3** **Core Barrel Assembly**

**Upper Flange Weld**

a. For one-sided visual inspections, the flaw is assumed to be through-wall.

b. Supplemental examinations may be used to determine depths of specific flaws as the basis for a flaw-specific evaluation, if needed.

3. Confirmation that fast neutron fluence at upper flange weld is below $3x10^{20}$ n/cm$^2$ (E> 1 MeV).

4. Steady-state normal operating stresses to be used to calculate SCC crack growth rates.

   a. Axial stresses based on normal mechanical loads are required for circumferential flaws.

   b. Stresses which have an insignificant net-through-wall value (average stress is near zero), such as weld residual stresses and thermal stresses due to local through-wall temperature gradients are considered to have minimal impact on the effective crack growth rates in through-wall flaws.

   c. Secondary weld residual and thermal stresses need to be considered in determination of circumferential and through-wall crack growth rates in partial through-wall flaws, whose dimensions would have to be determined with supplemental UT examinations.

5. Limiting externally applied transient stresses to be used to calculate allowable flaw lengths. Stresses arising from pressure, mechanical and thermal loads would be included in this calculation.

   a. More detailed load-deformation histories may be required for elastic-plastic or limit load calculations, if these calculations are necessary.

Analysis:

All analyses require an assumption of the SCC crack growth expected over the upcoming period of service. The methodology for analysis of a one-sided visual examination is based on an assumption of a through-wall flaw with weld residual and thermal stresses relieved. An appropriate crack growth rate model will be applied, which may be either K dependent or K independent. Fatigue crack growth has been assumed to be negligible. In order to apply the acceptance criteria to a period beyond one refueling cycle (not to exceed a 10-year re-inspection interval), additional action is required. The flaw evaluation will address the verification process of the predicted crack growth rate. Depending on the flaw size and knowledge of the plant conditions, a re-inspection at the next refueling outage may be required to provide data needed to operate beyond one cycle. The verification plan shall be included in the evaluation which is submitted to the regulator for their information. For detailed crack growth models used, see “Inputs and Assumptions” above.

Failure of the upper flange weld is assumed to occur when unstable circumferential crack growth is initiated from the analyzed flaw. Two options are outlined for determining the limiting allowable flaw length, based on neutron dose. Analysis methods are suggested for both pre-inspection or generic analysis (Suggested Pre-Inspection Analysis) and for flaws observed in-service



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**W-ID: 3** **Core Barrel Assembly**

## Upper Flange Weld

(Suggested Flaw Specific Analysis), where more detailed characteristics of the flaw and its location are known. In all cases, a more detailed evaluation may be completed using a semi-elliptic surface flaw, but such an evaluation would require more detailed inspection by UT.

| Fluence Range ($n/cm^2$ E>1MeV) | Dose (dpa) | Suggested Pre-Inspection Analysis | Suggested Flaw Specific Analysis |
|---|---|---|---|
| $\leq 3 \times 10^{20}$ | $\leq 0.5$ | LEFM using 150 ksi$\sqrt{in}$ as fracture toughness(1) or Limit Load | Limit Load |

Note:

1. Alternate fracture toughness values may be applied and justification for the alternate basis shall be submitted to the NRC for information within one year.

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

Option 1: LEFM Analysis

a. Establish initial crack length (and depth if determined by supplemental examinations) based on inspection results.
b. The final crack dimensions are calculated by adding 10 years of crack growth under normal loading conditions.
c. Ensure that the $\Delta K$ resulting from flow induced vibration (FIV) is below the threshold for fatigue crack growth.
d. For primary loading at normal and upset loading conditions, the final stress intensity factor (K) for the flaw size from (b) must be lower than the fracture toughness for the material by a factor of at least 2.77. For the evaluation of partial through wall flaws, the stress intensity factor due to secondary loading does not require an additional safety factor and may be directly added to the adjusted stress intensity factor for primary loading (i.e., $2.77 \cdot K_P + K_S < K_{IC}$).
e. For the governing primary loading at emergency or faulted loading conditions, the final stress intensity factor (K) for the flaw size from (b) must be lower than the fracture toughness for the material by a factor of at least 1.39. For the evaluation of partial through wall flaws, the stress intensity factor due to secondary loading does not require an additional safety factor and may be directly added to the adjusted stress intensity factor for primary loading (i.e., $1.39 \cdot K_P + K_S < K_{IC}$).



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**W-ID: 3** **Core Barrel Assembly**

**Upper Flange Weld**

Option 2: Limit Load Analysis (For neutron fluence < $3x10^{20}$ n/cm$^2$ at E>1 MeV or less than approximately 0.5 dpa)

a. Establish initial crack length (and depth if determined by supplemental examinations) based on inspection results.

b. The final crack dimensions are calculated by adding 10 years of crack growth under normal loading conditions.

c. Ensure that the ΔK resulting from flow induced vibration (FIV) is below the threshold for fatigue crack growth.

d. Determine the bending moment (M) that can be tolerated as a function of the postulated flaw length.

e. The applied moment, increased by a factor of 1.39 (for primary loading at emergency and faulted conditions) or 2.77 (for primary loading at normal and upset conditions) must be less than the limit moment (from step d) for the flaw length determined in (b), for the flaw to be acceptable. For the evaluation of partial through wall flaws, the moment due to secondary loading does not require an additional safety factor and may be directly added to the adjusted moment for primary loading (i.e., $[2.77 \text{ or } 1.39] \cdot M_p + M_s < M_{applied}$).

Acceptance Criteria: The upper flange weld continues to perform its functional requirements with the projected flaw length at the end of the inspection interval.

Approach: Limit load or LEFM analysis

Additional Action(s):

1) In order to apply the TR methodology for cracks detected via one-sided visual examinations, if supplementary examinations to determine crack depth were not performed, a follow up examination must be performed no later than the next RFO to confirm the assumed crack growth rates are conservative, unless a technical justification for a longer verification interval is submitted.

2) Note that the licensee shall submit the plant-specific crack growth and fracture mechanics analysis justifying the detected flaws, including a justification of the allowable operating period before re-inspection, within one year after any inspection that triggers the expansion criteria as defined in Table 5-3of MRP-227.



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| W-ID: 3.1 | Core Barrel Assembly<br>Upper Girth Weld | | |
|---|---|---|---|
| Category: | Expansion | Applicability: | All plants |
| Degradation Effect: | See MRP-227 Table 4-6 | | |
| Primary Link: | See MRP-227 Table 4-6 | | |
| Function: | Primary core support structure | | |

**Inspection**

| | |
|---|---|
| Method: | Visual examination. See MRP-227 Table 4-6. |
| Coverage: | See MRP-227 Table 4-6. |
| Observable Effect: | The specific relevant condition is a detectable crack-like surface indication. |

**Inputs and Assumptions**

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

- The inspections identified in MRP-227 are intended to provide a sampling of potential locations of degradation. Under this approach, inspection of one side (surface) of the weld is assumed to provide an adequate sampling for monitoring of SCC and/or Irradiation Assisted Stress Corrosion Cracking (IASCC).
- The change in resistance to fracture of the RV core barrel welds can be correlated to the accumulated fluence at each weld location. Welds that are subject to low fluence are considered to have a high degree of resistance to fracture. Correspondingly, those welds subject to high fluence have lower resistance to fracture.
- It is assumed that the primary mode of crack growth to be expected in this region is SCC/IASCC.
- The prediction of crack growth is based on the stress intensity factor, K, calculated using linear elastic fracture mechanics. The rate of crack growth is dependent on the amount of neutron fluence that the weld is expected to accumulate over the licensed operating lifetime. Since there has been very limited experience of IASCC initiated cracks in this or comparable components or welds in operating PWRs to date, growth rate models must be obtained from other sources. One source is the models developed for predicting IASCC crack growth rate (CGR) in BWR components. The other source is the models developed based on laboratory data that may include test results for both PWR and BWR representative conditions. These models were considered to be appropriate for the prediction of crack growth due to IASCC in PWR internals.
    - For weld locations subjected to fluence levels less than or equal to $5x10^{20}$ n/cm$^2$ (E>1MeV)(0.75 dpa), the maximum of the crack growth rate in 2017 ASME Section XI Appendix C-8520 **or** the crack growth equation in ASME Code Case N-889 [52] will be used.



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**W-ID: 3.1** **Core Barrel Assembly**

**Upper Girth Weld**

The conditions for the use of Code Case N-889 identified in Regulatory Guide RG 1.147, Rev. 20 [67] must be met.

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

- Depending on the magnitude of cyclic loads, such as thermal transients, additional crack growth as a result of these loads may need to be considered
- Acceptance criteria can be developed for the entire operating license of a given plant by using predicted end-of-license fluence values. However, re-inspection on the recommended intervals is required to confirm that the acceptance criteria are met. If there are changes to these fluence projections, such as in the event of a power uprate or change in core loading pattern, it would be necessary to confirm that the inputs selected based on fluence, such as SCC/IASCC growth rate and fracture toughness, remain applicable or bounding until the end of the operating license.

**Failure**

| | |
|---|---|
| Failure Mechanism: | Cracking (SCC, IASCC, fatigue) |
| Failure Effect: | Potential loss-of-core support. |
| Failure Criteria: | Actively growing through-wall flaws may require mitigation.<br>An existing flaw is unacceptable if the flaw length projected at the next inspection cycle allows a potential loss-of-core support. |

**Methodology**

| | |
|---|---|
| Goal: | Demonstrate that the cracking mechanism will not result in growth beyond the allowable crack length over the planned inspection interval. |
| Data Requirements: | 1. Surface crack length as determined by visual inspection<br>a. The 2013 edition of the ASME Code Section XI: IWA-3330(a) and Figure IWA-3330-1 provide requirements for the minimum allowable separation distance that will serve as the basis for guidance in this report. The requirement in Section XI is that the ligament between adjacent cracks must be greater than half of the thickness of the material. If this criterion is not met, the individual |



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**W-ID: 3.1** **Core Barrel Assembly**

**Upper Girth Weld**

crack lengths and the length of the ligament between the cracks must be summed. This total length is then compared to the allowable length. Note that the 2007 and 2010 editions of ASME Code Section XI have the equivalent content to the 2013 edition and are also applicable.

b. The flaw proximity rules in BWRVIP-158 [62] may be optionally applied in lieu of those defined in ASME Section XI: IWA-3330(a) and Figure IWA-3330-1 when appropriate.

2. Flaw Depth
   b. For one-sided visual inspections, the flaw is assumed to be through-wall.
   c. Supplemental examinations may be used to determine flaw depth, for a flaw-specific criterion, if needed.
3. Estimate fast neutron fluence at crack location.
4. Steady-state normal operating stresses to be used to calculate SCC/IASCC crack growth rates.
   a. Axial stresses based on normal mechanical loads are required for circumferential flaws.
   b. Stresses which have an insignificant net-through-wall value (average stress is near zero), such as weld residual stresses and thermal stresses due to local through-wall temperature gradients are considered to have minimal impact on the effective crack growth rates in through-wall flaws.
   c. Secondary weld residual and thermal stresses need to be considered in determination of circumferential and through-wall crack growth rates in partial through-wall flaws, whose dimensions would have to be determined with supplemental UT examinations.
5. Limiting externally applied transient stresses to be used to calculate allowable flaw lengths. Stresses arising from pressure, mechanical and thermal loads would be included in this calculation.
   a. More detailed load-deformation histories may be required for elastic-plastic or limit load calculations, if these calculations are necessary.

Analysis: All analyses require an assumption of the SCC/IASCC crack growth expected over the upcoming period of service. The methodology for analysis of a one-sided visual examination is based on an assumption of a through-wall flaw with weld residual and thermal stresses relieved. An appropriate crack growth rate model will be applied, which may be either K dependent or K independent. Fatigue crack growth has been assumed to be negligible. In order to apply the acceptance criteria to a period beyond one refueling cycle (not to exceed a 10-year re-inspection interval), additional action is required. The flaw evaluation will address the verification process of the predicted crack growth rate. Depending on the flaw size and knowledge of the plant conditions, a re-inspection at the next refueling outage may be required to provide data needed to operate beyond one cycle. The verification



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**W-ID: 3.1** **Core Barrel Assembly**

**Upper Girth Weld**

plan shall be included in the evaluation which is submitted to the regulator for their information. For detailed crack growth models used, see "Inputs and Assumptions" above.

Failure of the upper girth weld is assumed to occur when unstable circumferential crack growth is initiated from the analyzed flaw. Three options are outlined for determining the limiting allowable flaw length, based on neutron dose. Analysis methods are suggested for both pre-inspection or generic analysis (Suggested Pre-Inspection Analysis) and for flaws observed in-service (Suggested Flaw Specific Analysis), where more detailed characteristics of the flaw and its location are known. In all cases, a more detailed evaluation may be completed using a semi-elliptic surface flaw, but such an evaluation would require more detailed inspection by UT.

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

Note:

1. Alternate fracture toughness values may be applied and justification for the alternate basis shall be submitted to the NRC for information within one year.
2. MRP-211 [66] provides the basis for the fracture toughness value.

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



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**W-ID: 3.1** **Core Barrel Assembly**

**Upper Girth Weld**

Option 1: LEFM Analysis

a. Establish initial crack length (and depth if determined by supplemental examinations) based on inspection results.
b. The final crack dimensions are calculated by adding 10 years of crack growth under normal loading conditions.
c. Ensure that the ΔK resulting from flow induced vibration (FIV) is below the threshold for fatigue crack growth.
d. For primary loading at normal and upset loading conditions, the final stress intensity factor (K) for the flaw size from (b) must be lower than the fracture toughness for the material by a factor of at least 2.77. For the evaluation of partial through wall flaws, the stress intensity factor due to secondary loading does not require an additional safety factor and may be directly added to the adjusted stress intensity factor for primary loading (i.e., $2.77 \cdot K_P + K_S < K_{IC}$).
e. For the governing primary loading at emergency or faulted loading conditions, the final stress intensity factor (K) for the flaw size from (b) must be lower than the fracture toughness for the material by a factor of at least 1.39. For the evaluation of partial through wall flaws, the stress intensity factor due to secondary loading does not require an additional safety factor and may be directly added to the adjusted stress intensity factor for primary loading (i.e., $1.39 \cdot K_P + K_S < K_{IC}$).

Option 2: EPFM Analysis (for neutron dose levels > 0.5 dpa but < 5 dpa)

a. Establish initial crack length (and depth if determined by supplemental examinations) based on inspection results.
b. The final crack dimensions are calculated by adding 10 years of crack growth under normal loading conditions.
c. Ensure that the ΔK resulting from flow induced vibration (FIV) is below the threshold for fatigue crack growth.
d. Develop the (J/T) material curve from the material J-R curve.
e. The $J_{applied}$ curve must include a safety factor of 2.77 for primary loading at normal and upset conditions, and a factor of 1.39 for primary loading at emergency and faulted conditions. For the evaluation of partial through wall flaws, the J curve due to secondary loading does not require an additional safety factor and may be directly added to the adjusted J curve for primary loading (i.e., $[2.77 \text{ or } 1.39] \cdot J_p + J_s < J_{applied}$).
f. The intersection of the material and applied (J/T) curves indicates the instability point. The flaw size at instability is determined from the $J_{applied}$ versus flaw size curve.
g. The flaw size at instability must be larger than the flaw size from (b), for the flaw to be acceptable



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**W-ID: 3.1** **Core Barrel Assembly**

**Upper Girth Weld**

The intersection of the material and applied (J/T) curves indicates the instability point. The load at instability is determined from the Japplied versus load curve.

Option 3: Limit Load Analysis (For neutron fluence < $3x10^{20}$ n/cm$^2$ at E>1 MeV or dose less than approximately 0.5 dpa)

a. Establish initial crack length (and depth if determined by supplemental examinations) based on inspection results.

b. The final crack dimensions are calculated by adding 10 years of crack growth under normal loading conditions.

c. Ensure that the $\Delta K$ resulting from flow induced vibration (FIV) is below the threshold for fatigue crack growth.

d. Determine the bending moment (M) that can be tolerated as a function of the postulated flaw length.

e. The applied moment, increased by a factor of 1.39 (for primary loading at emergency and faulted conditions) or 2.77 (for primary loading at normal and upset conditions) must be less than the limit moment (from step d) for the flaw length determined in (b), for the flaw to be acceptable. For the evaluation of partial through wall flaws, the moment due to secondary loading does not require an additional safety factor and may be directly added to the adjusted moment for primary loading (i.e., $[2.77 \text{ or } 1.39] \cdot M_p + M_s < M_{applied}$).

Acceptance Criteria: The upper girth weld continues to perform their functional requirements with the projected flaw length at the end of the inspection interval.

Approach: Limit load, EPFM, or LEFM analysis

Additional Action(s):

1) In order to apply the TR methodology for cracks detected via one-sided visual examinations, if supplementary examinations to determine crack depth were not performed, a follow up examination must be performed no later than the next RFO to confirm the assumed crack growth rates are conservative, unless a technical justification for a longer verification interval is submitted.

2) Note that the licensee shall submit the plant-specific crack growth and fracture mechanics analysis justifying the detected flaws, including a justification of the allowable operating period before re-inspection, within one year after any inspection that detects relevant conditions as defined in Table 5-3 of MRP-227.



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| W-ID: 3.2 | Core Barrel Assembly<br>Upper Axial Weld | | |
|---|---|---|---|
| Category: | Expansion | Applicability: | All plants |
| Degradation Effect: | See MRP-227 Table 4-6 | | |
| Primary Link: | See MRP-227 Table 4-6 | | |
| Function: | The upper axial weld contributes to the overall structural integrity of the core barrel. | | |

**Inspection**

| | |
|---|---|
| Method: | Visual examination. See MRP-227 Table 4-6. |
| Coverage: | See MRP-227 Table 4-6 |
| Observable Effect: | The specific relevant condition is a detectable crack-like surface indication. |

**Inputs and Assumptions**

There are several inputs and assumptions that are critical to the development of acceptance criteria for the RV internals upper axial weld. These items are stated below:

- The inspections identified in MRP-227 are intended to provide a sampling of potential locations of degradation. Under this approach, inspection of one side (surface) of the weld is assumed to provide an adequate sampling for monitoring of SCC/IASCC.
- The change in resistance to fracture of the RV upper axial weld can be correlated to the accumulated fluence at each weld location. Welds that are subject to low fluence are considered to have a high degree of resistance to fracture. Correspondingly, those welds subject to high fluence have lower resistance to fracture.
- It is assumed that the primary mode of crack growth to be expected in this region is SCC/IASCC.
- The prediction of crack growth is based on the stress intensity factor, K, calculated using linear elastic fracture mechanics. The rate of crack growth is dependent on the amount of neutron fluence that the weld is expected to accumulate over the licensed operating lifetime. Since there has been very limited experience of IASCC initiated cracks in this or comparable components or welds in operating PWRs to date, growth rate models must be obtained from other sources. One source is the models developed for predicting IASCC crack growth rate (CGR) in BWR components. The other source is the models developed based on laboratory data that may include test results for both PWR and BWR representative conditions. These models were considered to be appropriate for the prediction of crack growth due to IASCC in PWR internals.
    - For weld locations subjected to fluence levels less than or equal to $5x10^{20}$ n/cm$^2$ (E>1MeV)(0.75 dpa), the maximum of the crack growth rate in 2017 ASME Section XI Appendix C-8520 **or** the



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**W-ID: 3.2** **Core Barrel Assembly**

**Upper Axial Weld**

crack growth equation in ASME Code Case N-889 [52] will be used. The conditions for the use of Code Case N-889 identified in Regulatory Guide RG 1.147, Rev. 20 [67] must be met.

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

- Depending on the magnitude of cyclic loads, such as thermal transients, additional crack growth as a result of these loads may need to be considered.
- Acceptance criteria can be developed for the entire operating license of a given plant by using predicted end-of-license fluence values. However, re-inspection on the recommended intervals is required to confirm that the acceptance criteria are met. If there are changes to these fluence projections, such as in the event of a power uprate or change in core loading pattern, it would be necessary to confirm that the inputs selected based on fluence, such as SCC growth rate and fracture toughness, remain applicable or bounding until the end of the operating license.

**Failure**

| | |
|---|---|
| Failure Mechanism: | Cracking (SCC, IASCC) |
| Failure Effect: | Potential for stress redistribution within the core barrel, which could initiate or accelerate circumferential crack growth in the core barrel. |
| Failure Criteria: | Actively growing through-wall flaws may require mitigation.<br>An existing flaw is unacceptable if the flaw length projected at the next inspection cycle results in a potential loss of core support. |

**Methodology**

| | |
|---|---|
| Goal: | Demonstrate that the cracking mechanism will not result in growth beyond the allowable crack length over the planned inspection interval. |
| Data Requirements: | 1. Surface crack length as determined by visual inspection<br>a. The 2013 edition of the ASME Code Section XI: IWA-3330(a) and Figure IWA-3330-1 provide requirements for the minimum allowable separation distance that will serve as the basis for |



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**W-ID: 3.2** **Core Barrel Assembly**

**Upper Axial Weld**

guidance in this report. The requirement in Section XI is that the ligament between adjacent cracks must be greater than half of the thickness of the material. If this criterion is not met, the individual crack lengths and the length of the ligament between the cracks must be summed. This total length is then compared to the allowable length. Note that the 2007 and 2010 editions of ASME Code Section XI have the equivalent content to the 2013 edition and are also applicable.

b. The flaw proximity rules in BWRVIP-158 [62] may be optionally applied in lieu of those defined in ASME Section XI: IWA-3330(a) and Figure IWA-3330-1 when appropriate.

2. Flaw Depth
   a. For one-sided visual inspections, the flaw is assumed to be through-wall.
   b. Supplemental examinations may be used to determine flaw depth, for a flaw-specific criterion, if needed.
3. Estimate fast neutron fluence at crack location.
4. Steady-state normal operating stresses to be used to calculate SCC crack growth rates.
   a. Hoop stresses based on normal mechanical loads are required for axial flaws.
   b. Stresses which have an insignificant net-through-wall value (average stress is near zero), such as weld residual stresses and thermal stresses due to local through-wall temperature gradients are considered to have minimal impact on the effective crack growth rates in through-wall flaws.
   c. Secondary weld residual and thermal stresses need to be considered in determination of axial and through-wall crack growth rates in partial through-wall flaws, whose dimensions would have to be determined with supplemental UT examinations.
5. Limiting externally applied transient stresses to be used to calculate allowable flaw lengths. Stresses arising from pressure, mechanical and thermal loads would be included in this calculation.
   a. More detailed load-deformation histories may be required for elastic-plastic or limit load calculations, if these calculations are necessary.



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**W-ID: 3.2** **Core Barrel Assembly**

**Upper Axial Weld**

Analysis: All analyses require an assumption of the SCC crack growth expected over the upcoming period of service. The methodology for analysis of a one-sided visual examination is based on an assumption of a through-wall flaw with weld residual and thermal stresses relieved. An appropriate crack growth rate model will be applied, which may be either K dependent or K independent. Fatigue crack growth has been assumed to be negligible. In order to apply the acceptance criteria to a period beyond one refueling cycle (not to exceed a 10-year re-inspection interval), additional action is required. The flaw evaluation will address the verification process of the predicted crack growth rate. Depending on the flaw size and knowledge of the plant conditions, a re-inspection at the next refueling outage may be required to provide data needed to operate beyond one cycle. The verification plan shall be included in the evaluation which is submitted to the regulator for their information. For detailed crack growth models used, see "Inputs and Assumptions" above.

Failure of the upper axial weld is assumed to occur when unstable axial crack growth is initiated from the analyzed flaw. Three options are outlined for determining the limiting allowable flaw length, based on neutron dose. Analysis methods are suggested for both pre-inspection or generic analysis (Suggested Pre-Inspection Analysis) and for flaws observed in-service (Suggested Flaw Specific Analysis), where more detailed characteristics of the flaw and its location are known. In all cases, a more detailed evaluation may be completed using a semi-elliptic surface flaw, but such an evaluation would require more detailed inspection by UT.



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**W-ID: 3.2** **Core Barrel Assembly**

**Upper Axial Weld**

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

Note:

1. Alternate fracture toughness values may be applied and justification for the alternate basis shall be submitted to the NRC for information within one year.
2. MRP-211 [66] provides the basis for the fracture toughness value.

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

Option 1: LEFM Analysis

a. Establish initial crack length (and depth if determined by supplemental examinations) based on inspection results.
b. The final crack dimensions are calculated by adding 10 years of crack growth under normal loading conditions.
c. Ensure that the $\Delta K$ resulting from flow-induced vibration (FIV) is below the threshold for fatigue crack growth.
d. For primary loading at normal and upset loading conditions, the final stress intensity factor (K) for the flaw size from (b) must be lower than the fracture toughness for the material by a factor of at least 2.77. For the evaluation of partial through wall flaws, the stress intensity factor due to secondary loading does not require an additional safety factor



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**W-ID: 3.2** **Core Barrel Assembly**

**Upper Axial Weld**

and may be directly added to the adjusted stress intensity factor for primary loading (i.e., $2.77 \cdot K_P + K_S < K_{IC}$).

e. For the governing primary loading at emergency or faulted loading conditions, the final stress intensity factor (K) for the flaw size from (b) must be lower than the fracture toughness for the material by a factor of at least 1.39. For the evaluation of partial through wall flaws, the stress intensity factor due to secondary loading does not require an additional safety factor and may be directly added to the adjusted stress intensity factor for primary loading (i.e., $1.39 \cdot K_P + K_S < K_{IC}$).

Option 2: EPFM Analysis (for neutron dose levels > 0.5 dpa but < 5 dpa)

a. Establish initial crack length (and depth if determined by supplemental examinations) based on inspection results.

b. The final crack dimensions are calculated by adding 10 years of crack growth under normal loading conditions.

c. Ensure that the ΔK resulting from flow-induced vibration (FIV) is below the threshold for fatigue crack growth.

d. Develop the (J/T) material curve from the material J-R curve.

e. The $J_{applied}$ curve must include a safety factor of 2.77 for primary loading at normal and upset conditions, and a factor of 1.39 for primary loading at emergency and faulted conditions. For the evaluation of partial through wall flaws, the J curve due to secondary loading does not require an additional safety factor and may be directly added to the adjusted J curve for primary loading (i.e., $[2.77 \text{ or } 1.39] \cdot J_p + J_s < J_{applied}$).

f. The intersection of the material and applied (J/T) curves indicates the instability point. The flaw size at instability is determined from the $J_{applied}$ versus flaw size curve.

g. The flaw size at instability must be larger than the flaw size from (b), for the flaw to be acceptable.

The intersection of the material and applied (J/T) curves indicates the instability point.

The load at instability is determined from the $J_{applied}$ versus load curve.

Option 3: Limit Load Analysis (For neutron fluence < $3 \times 10^{20}$ n/cm$^2$ at E>1 MeV or dose less than approximately 0.5 dpa)

a. Establish initial crack length (and depth if determined by supplemental examinations) based on inspection results.

b. The final crack dimensions are calculated by adding 10 years of crack growth under normal loading conditions.

c. Ensure that the ΔK resulting from flow-induced vibration (FIV) is below the threshold for fatigue crack growth.



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**W-ID: 3.2** **Core Barrel Assembly**

**Upper Axial Weld**

d. Determine the bending moment (M) that can be tolerated as a function of the postulated flaw length.

e. The applied moment, increased by a factor of 1.39 (for primary loading at emergency and faulted conditions) or 2.77 (for primary loading at normal and upset conditions) must be less than the limit moment (from step d) for the flaw length determined in (b), for the flaw to be acceptable. For the evaluation of partial through wall flaws, the moment due to secondary loading does not require an additional safety factor and may be directly added to the adjusted moment for primary loading (i.e., $[2.77 \text{ or } 1.39] \cdot M_p + M_s < M_{applied}$).

Acceptance Criteria: The upper axial weld continues to perform their functional requirements with the projected flaw lengths at the end of the inspection interval.

Approach: Limit load, EPFM, or LEFM analysis

Additional Action(s):

1) In order to apply the TR methodology for cracks detected via one-sided visual examinations, if supplementary examinations to determine crack depth were not performed, a follow up examination must be performed no later than the next RFO to confirm the assumed crack growth rates are conservative, unless a technical justification for a longer verification interval is submitted.

2) Note that the licensee shall submit the plant-specific crack growth and fracture mechanics analysis justifying the detected flaws, including a justification of the allowable operating period before re-inspection, within one year after any inspection that detects relevant conditions as defined in Table 5-3 of MRP-227.



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| W-ID: 3.3 | Core Barrel Assembly<br>Lower Flange Weld | | |
|---|---|---|---|
| Category: | Expansion | Applicability: | All plants |
| Degradation Effect: | See MRP-227 Table 4-6 | | |
| Primary Link: | See MRP-227 Table 4-6 | | |
| Function: | Primary core support structure | | |

**Inspection**

| | |
|---|---|
| Method: | Visual examination. See MRP-227 Table 4-6. |
| Coverage: | See MRP-227 Table 4-6 |
| Observable Effect: | The specific relevant condition is a detectable crack-like surface indication. |

**Inputs and Assumptions**

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

- The inspections identified in MRP-227 are intended to provide a sampling of potential locations of degradation. Under this approach, inspection of one side (surface) of the weld is assumed to provide an adequate sampling for monitoring of SCC and fatigue.
- It is assumed that these welds experience low fluence. The change in resistance to fracture of the RV lower flange weld can be correlated to the accumulated fluence at each weld location. Welds that are subject to low fluence are considered to have a high degree of resistance to fracture. Correspondingly, those welds subject to high fluence have lower resistance to fracture.
- It is assumed that the primary mode of crack growth to be expected in this region is stress corrosion cracking.
- The prediction of crack growth is based on the stress intensity factor, K, calculated using linear elastic fracture mechanics. The rate of crack growth is dependent on the amount of neutron fluence that the weld is expected to accumulate over the licensed operating lifetime. Since there has been very limited experience of SCC initiated cracks in operating PWRs to date, growth rates developed for the prediction of SCC in boiling water reactors (BWRs) are assumed to be appropriate for prediction of crack growth due to SCC in PWR reactor internals.
    - For weld locations subjected to fluence levels less than or equal to $5x10^{20}$ n/cm$^2$ (E>1MeV)(0.75 dpa), the maximum of the crack growth rate in 2017 ASME Section XI Appendix C-8520 **or** the crack growth equation in ASME Code Case N-889 [52] will be used. The conditions for the use of Code Case N-889 identified in Regulatory Guide RG 1.147, Rev. 20 [67] must be met.
    - For fluence levels above $5x10^{20}$ n/cm$^2$ (E>1MeV)(0.75 dpa) up to and including $1.4x10^{22}$ n/cm$^2$ (E>1 MeV) (20 dpa), the crack



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**W-ID: 3.3** **Core Barrel Assembly**

**Lower Flange Weld**

growth equation in ASME Code Case N-889 [52] will be used. The conditions for the use of Code Case N-889 identified in Regulatory Guide RG 1.147, Rev. 20 [67] must be met for the development of acceptance criteria.

- Depending on the magnitude of cyclic loads, such as thermal transients, additional crack growth as a result of these loads may need to be considered.
- Acceptance criteria can be developed for the entire operating license of a given plant by using predicted end-of-license fluence values. However, re-inspection on the recommended intervals is required to confirm that the acceptance criteria are met. If there are changes to these fluence projections, such as in the event of a power uprate or change in core loading pattern, it would be necessary to confirm that the inputs selected based on fluence, such as SCC growth rate and fracture toughness, remain applicable or bounding until the end of the operating license.

**Failure**

| | |
|---|---|
| Failure Mechanism: | Cracking (SCC, Fatigue) |
| Failure Effect: | Potential loss-of-core support. |
| Failure Criteria: | Actively growing through-wall flaws may require mitigation.<br>An existing flaw is unacceptable if the flaw length projected at the next inspection cycle allows a potential loss-of-core support. |

**Methodology**

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

Data Requirements:

1. Surface crack length as determined by visual inspection
   a. The 2013 edition of the ASME Code Section XI: IWA-3330(a) and Figure IWA-3330-1 provide requirements for the minimum allowable separation distance that will serve as the basis for guidance in this report. The requirement in Section XI is that the ligament between adjacent cracks must be greater than half of the thickness of the material. If this criterion is not met, the individual crack lengths and the length of the ligament between the cracks must be summed. This total length is then compared to the allowable length. Note that the 2007 and 2010 editions of ASME Code Section XI have the equivalent content to the 2013 edition and are also applicable.
   b. The flaw proximity rules in BWRVIP-158 [62] may be optionally applied in lieu of those defined in ASME Section XI: IWA-3330(a) and Figure IWA-3330-1 when appropriate.
2. Flaw Depth



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**W-ID: 3.3** **Core Barrel Assembly**

**Lower Flange Weld**

a. For one-sided visual inspections, the flaw is assumed to be through-wall.

b. Supplemental examinations may be used to determine flaw depth, for a flaw-specific criterion, if needed.

3. Estimate of fast neutron fluence at crack location.
4. Steady-state normal operating stresses to be used to calculate SCC crack growth rates.
    a. Axial stresses based on normal mechanical loads are required for circumferential flaws.
    b. Stresses which have an insignificant net-through-wall value (average stress is near zero), such as weld residual stresses and thermal stresses due to local through-wall temperature gradients are considered to have minimal impact on the effective crack growth rates in through-wall flaws.
    c. Secondary weld residual and thermal stresses need to be considered in determination of circumferential and through-wall crack growth rates in partial through-wall flaws, whose dimensions would have to be determined with supplemental UT examinations.
5. Limiting externally applied transient stresses to be used to calculate allowable flaw lengths. Stresses arising from pressure, mechanical and thermal loads would be included in this calculation.
    a. More detailed load-deformation histories may be required for elastic-plastic or limit load calculations, if these calculations are necessary.

Analysis:

All analyses require an assumption of the SCC crack growth expected over the upcoming period of service. The methodology for analysis of a one-sided visual examination is based on an assumption of a through-wall flaw with weld residual and thermal stresses relieved. An appropriate crack growth rate model will be applied, which may be either K dependent or K independent. Fatigue crack growth has been assumed to be negligible. In order to apply the acceptance criteria to a period beyond one refueling cycle (not to exceed a 10-year re-inspection interval), additional action is required. The flaw evaluation will address the verification process of the predicted crack growth rate. Depending on the flaw size and knowledge of the plant conditions, a re-inspection at the next refueling outage may be required to provide data needed to operate beyond one cycle. The verification plan shall be included in the evaluation which is submitted to the regulator for their information. For detailed crack growth models used, see "Inputs and Assumptions" above.

Failure of the lower flange weld is assumed to occur when unstable circumferential crack growth is initiated from the analyzed flaw. Two options are outlined for determining the limiting allowable flaw length, based on neutron dose. Analysis methods are suggested for both pre-inspection and generic analysis (Suggested Pre-Inspection Analysis) and for flaws observed in-service



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## W-ID: 3.3 Core Barrel Assembly

### Lower Flange Weld

(Suggested Flaw Specific Analysis), where more detailed characteristics of the flaw and its location are known. In all cases, a more detailed evaluation may be completed using a semi-elliptic surface flaw, but such an evaluation would require more detailed inspection by UT.

| Fluence Range ($n/cm^2$ E>1MeV) | Dose (dpa) | Suggested Pre-Inspection Analysis | Suggested Flaw Specific Analysis |
|---|---|---|---|
| $\leq 3 \times 10^{20}$ | $\leq 0.5$ | LEFM using 150 ksi$\sqrt{in}$ as fracture toughness(1) or Limit Load | Limit Load |

Note:

1. Alternate fracture toughness values may be applied and justification for the alternate basis shall be submitted to the NRC for information within one year.

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

Option 1: LEFM Analysis

a. Establish initial crack length (and depth if determined by supplemental examinations) based on inspection results.
b. The final crack dimensions are calculated by adding 10 years of crack growth under normal loading conditions.
c. Ensure that the $\Delta K$ resulting from flow-induced vibration (FIV) is below the threshold for fatigue crack growth.
d. For primary loading at normal and upset loading conditions, the final stress intensity factor (K) for the flaw size from (b) must be lower than the fracture toughness for the material by a factor of at least 2.77. For the evaluation of partial through wall flaws, the stress intensity factor due to secondary loading does not require an additional safety factor and may be directly added to the adjusted stress intensity factor for primary loading (i.e., $2.77 \cdot K_P + K_S < K_{IC}$).
e. For the governing primary loading at emergency or faulted loading conditions, the final stress intensity factor (K) for the flaw size from (b) must be lower than the fracture toughness for the material by a factor of at least 1.39. For the evaluation of partial through wall flaws, the stress intensity factor due to secondary loading does not require an additional safety factor and may be directly added to the adjusted stress intensity factor for primary loading (i.e., $1.39 \cdot K_P + K_S < K_{IC}$).



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**W-ID: 3.3** **Core Barrel Assembly**

**Lower Flange Weld**

Option 2: Limit Load Analysis (For neutron fluence < $3x10^{20}$ n/cm$^2$ at E>1 MeV or dose less than approximately 0.5 dpa)

a. Establish initial crack length (and depth if determined by supplemental examinations) based on inspection results.

b. The final crack dimensions are calculated by adding 10 years of crack growth under normal loading conditions.

c. Ensure that the $\Delta K$ resulting from flow-induced vibration (FIV) is below the threshold for fatigue crack growth.

d. Determine the bending moment (M) that can be tolerated as a function of the postulated flaw length.

e. The applied moment, increased by a factor of 1.39 (for primary loading at emergency and faulted conditions) or 2.77 (for primary loading at normal and upset conditions) must be less than the limit moment (from step d) for the flaw length determined in (b), for the flaw to be acceptable. For the evaluation of partial through wall flaws, the moment due to secondary loading does not require an additional safety factor and may be directly added to the adjusted moment for primary loading (i.e., $[2.77 \text{ or } 1.39] \cdot M_p + M_s < M_{applied}$).

Acceptance Criteria: The lower flange weld continues to perform its functional requirements with the projected flaw length at the end of the interval.

Approach: Limit load or LEFM analysis

Additional Action(s):

1) In order to apply the TR methodology for cracks detected via one-sided visual examinations, if supplementary examinations to determine crack depth were not performed, a follow up examination must be performed no later than the next RFO to confirm the assumed crack growth rates are conservative, unless a technical justification for a longer verification interval is submitted.

2) Note that the licensee shall submit the plant-specific crack growth and fracture mechanics analysis justifying the detected flaws, including a justification of the allowable operating period before re-inspection, within one year after any inspection that detects relevant conditions as defined in Table 5-3 of MRP-227.



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| W-ID: 3.4 | **Lower Internals Assembly**<br>**Lower Support Forging or Casting** |
|---|---|
| Category: | Expansion Applicability: All plants |
| Degradation Effect: | See MRP-227 Table 4-6 |
| Primary Link: | See MRP-227 Table 4-6 |
| Function: | Primary core support structure |

**Inspection**

| | |
|---|---|
| Method: | Visual examination. See MRP-227 Table 4-6. |
| Coverage: | See MRP-227 Table 4-6 |
| Observable Effect: | The specific relevant condition is a detectable crack-like surface indication. |

**Inputs and Assumptions**

The acceptance criteria methodology assumes that all other components in the lower internals assembly are in a normal, non-degraded condition. If additional flaws are found in the lower internals assembly, additional analysis of the as-found condition is required to determine suitability for return to service.

Assume crack initiates at the location with the highest surface tensile stress.

Assume the crack propagates through the full thickness of the forging or casting through the ligament with the highest stress to the adjacent hole.

**Failure**

| | |
|---|---|
| Failure Mechanism: | Cracking, Aging Management (Thermal Embrittlement in Casting) |
| Failure Effect: | Potential loss-of-core support. |
| Failure Criteria: | An existing flaw is unacceptable if the flaw length projected at the next inspection cycle allows a potential loss-of-core support. |

**Methodology**

Goal: Determine the level of cracking required to weaken the lower support forging or casting to the point that it cannot adequately support the lower core plate.

Data Requirements:
Operating loads
Functional requirements for the lower support forging or casting and the lower core plate.

Analysis: Determine acceptable flaw pattern.

1. Establish functional requirements for the lower core plate assembly
   a. During normal operation, the system of support columns resists lower core plate deflection caused by mechanical and thermal loading. The lower core plate has requirements for “flatness” affecting fuel assembly alignment.



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**W-ID: 3.4** **Lower Internals Assembly**

**Lower Support Forging or Casting**

b. During the limiting transient, the system must maintain structural integrity (consult design specification for flatness requirements during transients)

2. Structural model of lower support structure
   a. Develop a model of the lower support structure that includes support columns, lower forging or casting, and lower core plate. The deflected shape of the lower support forging or casting will influence the deflection of the lower core plate through the support columns which join them. The model should be capable of modeling a crack connecting holes in the lower core support forging or casting (crack tip modeling not required). Sector modeling based on symmetry may be justifiable. Evaluate the lower core plate against the functional requirements (e.g. deflection of the plate and the alignment of the guide pins).
3. Any single observed crack is acceptable if the FEA model predicts that the functional requirement s of the lower core plate (e.g. deflection of the plate and the alignment of the guide pins) are met.
4. Determine that the observed crack does not cause stresses in the remainder of the lower core support forging or casting to exceed applicable limits.
5. If unable to demonstrate the acceptability of a single crack, a detailed flaw analysis is required.
6. Optional determination of margin for additional cracking: Repeat the evaluation for multiple cracks connecting adjacent holes. Determine the number and pattern of connected holes which causes violation of the functional requirements.

Acceptance Criteria: Current cracking does not cause deflections which violate lower core plate assembly functional requirements or produce loose parts. Analysis does not predict development of additional cracks in excess of the acceptable flaw pattern during the inspection interval.

Note: Once cracking is observed, reinspection is required during each refueling outage unless additional analysis can justify a longer inspection interval.

Approach: FEA (plant specific)

Additional Action(s): Note that the licensee shall submit the plant-specific analysis for this component within one year after any inspection that detects relevant conditions as defined in Table 5-3 of MRP-227.



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| W-ID: 4 | Core Barrel Assembly<br>Lower Girth Weld | | |
|---|---|---|---|
| Category: | Primary | Applicability: | All plants |
| Degradation Effect: | See MRP-227 Table 4-3 | | |
| Expansion Link: | See MRP-227 Table 4-3 | | |
| Function: | Primary core support structure | | |

**Inspection**

| | |
|---|---|
| Method: | Visual examination. See MRP-227 Table 4-3. |
| Coverage: | See MRP-227 Table 4-3 |
| Observable Effect: | The specific relevant condition is a detectable crack-like surface indication. |

**Inputs and Assumptions**

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

- The inspections identified in MRP-227 are intended to provide a sampling of potential locations of degradation. Under this approach, inspection of one side (surface) of the weld is assumed to provide an adequate sampling for monitoring of SCC and/or Irradiation Assisted Stress Corrosion Cracking (IASCC).
- The change in resistance to fracture of the RV core barrel welds can be correlated to the accumulated fluence at each weld location. Welds that are subject to low fluence are considered to have a high degree of resistance to fracture. Correspondingly, those welds subject to high fluence have lower resistance to fracture.
- It is assumed that the primary mode of crack growth to be expected in this region is SCC/IASCC.
- The prediction of crack growth is based on the stress intensity factor, K, calculated using linear elastic fracture mechanics. The rate of crack growth is dependent on the amount of neutron fluence that the weld is expected to accumulate over the licensed operating lifetime. Since there has been very limited experience of IASCC initiated cracks in this or comparable components or welds in operating PWRs to date, growth rate models must be obtained from other sources. One source is the models developed for predicting IASCC crack growth rate (CGR) in BWR components. The other source is the models developed based on laboratory data that may include test results for both PWR and BWR representative conditions. These models were considered to be appropriate for the prediction of crack growth due to IASCC in PWR internals.
    - For weld locations subjected to fluence levels less than or equal to $5x10^{20}$ n/cm$^2$ (E>1MeV)(0.75 dpa), the maximum of the crack growth rate in 2017 ASME Section XI Appendix C-8520 **or** the crack growth equation in ASME Code Case N-889 [52] will be



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**W-ID: 4** **Core Barrel Assembly**

**Lower Girth Weld**

used. The conditions for the use of Code Case N-889 identified in Regulatory Guide RG 1.147, Rev. 20 [67] must be met.

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

- Depending on the magnitude of cyclic loads, such as thermal transients, additional crack growth as a result of these loads may need to be considered
- Acceptance criteria can be developed for the entire operating license of a given plant by using predicted end-of-license fluence values. However, re-inspection on the recommended intervals is required to confirm that the acceptance criteria are met. If there are changes to these fluence projections, such as in the event of a power uprate or change in core loading pattern, it would be necessary to confirm that the inputs selected based on fluence, such as SCC/IASCC growth rate and fracture toughness, remain applicable or bounding until the end of the operating license.

**Failure**

| | |
|---|---|
| Failure Mechanism: | Cracking (SCC, IASCC, fatigue) |
| Failure Effect: | Potential loss-of-core support. |
| Failure Criteria: | Actively growing through-wall flaws may require mitigation.<br>An existing flaw is unacceptable if the flaw length projected at the next inspection cycle allows a potential loss-of-core support. |

**Methodology**

| | |
|---|---|
| Goal: | Demonstrate that the cracking mechanism will not result in growth beyond the allowable crack length over the planned inspection interval. |
| Data Requirements: | 1. Surface crack length as determined by visual inspection<br>a. The 2013 edition of the ASME Code Section XI: IWA-3330(a) and Figure IWA-3330-1 provide requirements for the minimum allowable separation distance that will serve as the basis for guidance in this report. The requirement in Section XI is that the ligament between adjacent cracks must be greater than half of the |



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**W-ID: 4** **Core Barrel Assembly**

**Lower Girth Weld**

thickness of the material. If this criterion is not met, the individual crack lengths and the length of the ligament between the cracks must be summed. This total length is then compared to the allowable length. Note that the 2007 and 2010 editions of ASME Code Section XI have the equivalent content to the 2013 edition and are also applicable.

b. The flaw proximity rules in BWRVIP-158 [62] may be optionally applied in lieu of those defined in ASME Section XI: IWA-3330(a) and Figure IWA-3330-1 when appropriate..

2. Flaw Depth
   a. For one-sided visual inspections, the flaw is assumed to be through-wall.
   b. Supplemental examinations may be used to determine flaw depth, for a flaw-specific criterion, if needed.
3. Estimate fast neutron fluence at crack location.
4. Steady-state normal operating stresses to be used to calculate SCC/IASCC crack growth rates.
   a. Axial stresses based on normal mechanical loads are required for circumferential flaws.
   b. Stresses which have an insignificant net-through-wall value (average stress is near zero), such as weld residual stresses and thermal stresses due to local through-wall temperature gradients are considered to have minimal impact on the effective crack growth rates in through-wall flaws.
   c. Secondary weld residual and thermal stresses need to be considered in determination of circumferential and through-wall crack growth rates in partial through-wall flaws, whose dimensions would have to be determined with supplemental UT examinations.
5. Limiting externally applied transient stresses to be used to calculate allowable flaw lengths. Stresses arising from pressure, mechanical and thermal loads would be included in this calculation.
   a. More detailed load-deformation histories may be required for elastic-plastic or limit load calculations, if these calculations are necessary.

Analysis: All analyses require an assumption of the SCC/IASCC crack growth expected over the upcoming period of service. The methodology for analysis of a one-sided visual examination is based on an assumption of a through-wall flaw with weld residual and thermal stresses relieved. An appropriate crack growth rate model will be applied, which may be either K dependent or K independent. Fatigue crack growth has been assumed to be negligible. In order to apply the acceptance criteria to a period beyond one refueling cycle (not to exceed a 10-year re-inspection interval), additional action is required. The flaw evaluation will address the verification process of the predicted crack growth



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**Lower Girth Weld**

rate. Depending on the flaw size and knowledge of the plant conditions, a re-inspection at the next refueling outage may be required to provide data needed to operate beyond one cycle. The verification plan shall be included in the evaluation which is submitted to the regulator for their information. For detailed crack growth models used, see "Inputs and Assumptions" above.

Failure of the lower girth weld is assumed to occur when unstable circumferential crack growth is initiated from the analyzed flaw. Three options are outlined for determining the limiting allowable flaw length, based on neutron dose. Analysis methods are suggested for both pre-inspection or generic analysis (Suggested Pre-Inspection Analysis) and for flaws observed in-service (Suggested Flaw Specific Analysis), where more detailed characteristics of the flaw and its location are known. In all cases, a more detailed evaluation may be completed using a semi-elliptic surface flaw, but such an evaluation would require more detailed inspection by UT.

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

Note:

1. Alternate fracture toughness values may be applied and justification for the alternate basis shall be submitted to the NRC for information within one year.
2. MRP-211 [66] provides the basis for the fracture toughness value.

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

Option 1: LEFM Analysis



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**Lower Girth Weld**

a. Establish initial crack length (and depth if determined by supplemental examinations) based on inspection results.

b. The final crack dimensions are calculated by adding 10 years of crack growth under normal loading conditions.

c. Ensure that the ΔK resulting from flow induced vibration (FIV) is below the threshold for fatigue crack growth.

d. For primary loading at normal and upset loading conditions, the final stress intensity factor (K) for the flaw size from (b) must be lower than the fracture toughness for the material by a factor of at least 2.77. For the evaluation of partial through wall flaws, the stress intensity factor due to secondary loading does not require an additional safety factor and may be directly added to the adjusted stress intensity factor for primary loading (i.e., $2.77 \cdot K_P + K_S < K_{IC}$).

e. For the governing primary loading at emergency or faulted loading conditions, the final stress intensity factor (K) for the flaw size from (b) must be lower than the fracture toughness for the material by a factor of at least 1.39. For the evaluation of partial through wall flaws, the stress intensity factor due to secondary loading does not require an additional safety factor and may be directly added to the adjusted stress intensity factor for primary loading (i.e., $1.39 \cdot K_P + K_S < K_{IC}$).

Option 2: EPFM Analysis (for neutron dose levels > 0.5 dpa but < 5 dpa)

a. Establish initial crack length (and depth if determined by supplemental examinations) based on inspection results.

b. The final crack dimensions are calculated by adding 10 years of crack growth under normal loading conditions.

c. Ensure that the ΔK resulting from flow induced vibration (FIV) is below the threshold for fatigue crack growth.

d. Develop the (J/T) material curve from the material J-R curve.

e. The $J_{applied}$ curve must include a safety factor of 2.77 for primary loading at normal and upset conditions, and a factor of 1.39 for primary loading at emergency and faulted conditions. For the evaluation of partial through wall flaws, the J curve due to secondary loading does not require an additional safety factor and may be directly added to the adjusted J curve for primary loading (i.e., $[2.77 \text{ or } 1.39] \cdot J_p + J_s < J_{applied}$).

f. The intersection of the material and applied (J/T) curves indicates the instability point. The flaw size at instability is determined from the $J_{applied}$ versus flaw size curve.

g. The flaw size at instability must be larger than the flaw size from (b), for the flaw to be acceptable

The intersection of the material and applied (J/T) curves indicates the instability point. The load at instability is determined from the Japplied versus load curve.



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**W-ID: 4** **Core Barrel Assembly**

**Lower Girth Weld**

Option 3: Limit Load Analysis (For neutron fluence < $3x10^{20}$ n/cm$^2$ at E>1 MeV or dose less than approximately 0.5 dpa)

a. Establish initial crack length (and depth if determined by supplemental examinations) based on inspection results.

b. The final crack dimensions are calculated by adding 10 years of crack growth under normal loading conditions.

c. Ensure that the $\Delta K$ resulting from flow induced vibration (FIV) is below the threshold for fatigue crack growth.

d. Determine the bending moment (M) that can be tolerated as a function of the postulated flaw length.

e. The applied moment, increased by a factor of 1.39 (for primary loading at emergency and faulted conditions) or 2.77 (for primary loading at normal and upset conditions) must be less than the limit moment (from step d) for the flaw length determined in (b), for the flaw to be acceptable. For the evaluation of partial through wall flaws, the moment due to secondary loading does not require an additional safety factor and may be directly added to the adjusted moment for primary loading (i.e., $[2.77 \text{ or } 1.39] \cdot M_p + M_s < M_{applied}$).

Acceptance Criteria: The lower girth weld continues to perform their functional requirements with the projected flaw length at the end of the inspection interval.

Approach: Limit load, EPFM, or LEFM analysis

Additional Action(s):

1) In order to apply the TR methodology for cracks detected via one-sided visual examinations, if supplementary examinations to determine crack depth were not performed, a follow up examination must be performed no later than the next RFO to confirm the assumed crack growth rates are conservative, unless a technical justification for a longer verification interval is submitted.

2) Note that the licensee shall submit the plant-specific crack growth and fracture mechanics analysis justifying the detected flaws, including a justification of the allowable operating period before re-inspection, within one year after any inspection that triggers the expansion criteria as defined in Table 5-3 of MRP-227.



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| W-ID: 4.1 | Upper Internals Assembly<br>Upper Core Plate | | |
|---|---|---|---|
| Category: | Expansion | Applicability: | All plants |
| Degradation Effect: | See MRP-227 Table 4-6 | | |
| Primary Link: | See MRP-227 Table 4-6 | | |
| Function: | Primary core support structure | | |

**Inspection**

| | |
|---|---|
| Method: | Visual examination. See MRP-227 Table 4-6. |
| Coverage: | See MRP-227 Table 4-6 |
| Observable Effect: | The specific relevant condition is a detectable crack-like surface indication. |

**Inputs and Assumptions**

The acceptance criteria methodology assumes that cracking due to fatigue is the active degradation mechanism and that all other components in the upper internals assembly are in a nominal, non-degraded condition. If additional flaws are found in the upper internals assembly, additional analysis of the as-found assembly is required to determine suitability for return to service.

**Failure**

| | |
|---|---|
| Failure Mechanism: | Cracking (fatigue), wear |
| Failure Effect: | 1. Loss of core support<br>2. Difficulty in loading, or removing, fuel due to misalignment of the guide pins. |
| Failure Criteria: | Deflection of the upper core plate with accompanying misalignment of the fuel assembly guide pins, upper core plate alignment pin inserts, or guide tubes beyond the maximum permissible for proper function. |

**Methodology**

| | |
|---|---|
| Goal: | Cracks in the upper core plate are expected to grow from hole-to-hole within the plate. A network of connected cracks is required to allow significant degradation of the plate functionality. The goal is to demonstrate that the cracking which might be present does not violate the functional requirements. |
| Data Requirements: | 1. Operating loads (e.g., pressure, flow, and thermal for limiting conditions)<br>2. Functional requirements for the upper core plate assembly. |
| Analysis: | Determine acceptable flaw pattern.<br>1. Establish functional requirements for the upper core plate assembly.<br>a. During normal operation, the system of support columns resists upper core plate deflection caused by mechanical and thermal loading. The upper core plate requirements may include maximum allowable deflection affecting guide tube and fuel assembly alignment. The upper core plate alignment pins and inserts affect fuel assembly lateral |



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**W-ID: 4.1** **Upper Internals Assembly**

**Upper Core Plate**

support during earthquakes, LOCA, and other transients. Requirements for these components may also need to be considered.

b. During the limiting accident transient the system must maintain structural integrity.

2. Upper core plate analysis assumptions.
   a. Assume crack initiates in the plate at the location with the highest surface tensile stress.
   b. Assume the crack propagates through the full thickness of the plate through the ligament with highest stress to the adjacent hole.
3. Structural model of upper support structure.

   Develop a model of the upper support structure that may include support columns, upper support weldment, guide tubes, upper core plate alignment pins and inserts, and upper core plate. The model should be capable of modeling a crack connecting holes in the upper core plate (crack tip modeling not required). Sector modeling based on symmetry may be justifiable. Evaluate the resulting deflection of the upper core plate and the alignment of the guide tubes, guide pins and alignment pins/inserts against the functional requirements.
4. Any single observed crack is acceptable if displacement of the upper core plate and alignment of the guide pins and alignment pins/inserts in the FEA model meets the functional requirements.
5. Determine that the observed crack does not cause stresses in the remainder of the upper core plate to exceed applicable limits.
6. If unable to demonstrate the acceptability of a single crack, a detailed flaw analysis is required.
7. Optional determination of margin for additional cracking: Repeat the evaluation for multiple cracks connecting adjacent holes. Determine the number and pattern of connected holes which causes violation of the functional requirements.

Acceptance Criteria: The observed cracking does not cause deflections which violate the upper core plate assembly functional requirements. Fatigue analysis does not predict development of additional cracks in excess of the acceptable flaw pattern during the inspection interval.

Note: Once cracking is observed, re-inspection is required during each subsequent refueling outage unless additional analysis can justify a longer inspection interval.

Approach: FEA (plant specific)

Additional Action(s): Note that the licensee shall submit the plant-specific analysis for this component within one year after any inspection that detects relevant conditions as defined in Table 5-3 of MRP-227.



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| W-ID: 4.2 | Core Barrel Assembly<br>Middle Axial Welds | | |
|---|---|---|---|
| Category: | Expansion | Applicability: | All plants |
| Degradation Effect: | See MRP-227 Table 4-6 | | |
| Primary Link: | See MRP-227 Table 4-6 | | |
| Function: | The middle axial welds contribute to the overall structural integrity of the core barrel. | | |

**Inspection**

| | |
|---|---|
| Method: | Visual examination. See MRP-227 Table 4-6. |
| Coverage: | See MRP-227 Table 4-6. |
| Observable Effect: | The specific relevant condition is a detectable crack-like surface indication. |

**Inputs and Assumptions**

There are several inputs and assumptions that are critical to the development of acceptance criteria for the RV internals middle axial welds. These items are stated below:

- The inspections identified in MRP-227 are intended to provide a sampling of potential locations of degradation. Under this approach, inspection of one side (surface) of the weld is assumed to provide an adequate sampling for monitoring of SCC/IASCC.
- The change in resistance to fracture of the RV middle axial welds can be correlated to the accumulated fluence at each weld location. Welds that are subject to low fluence are considered to have a high degree of resistance to fracture. Correspondingly, those welds subject to high fluence have lower resistance to fracture.
- It is assumed that the primary mode of crack growth to be expected in this region is SCC/IASCC.
- The prediction of crack growth is based on the stress intensity factor, K, calculated using linear elastic fracture mechanics. The rate of crack growth is dependent on the amount of neutron fluence that the weld is expected to accumulate over the licensed operating lifetime. Since there has been very limited experience of IASCC initiated cracks in this or comparable components or welds in operating PWRs to date, growth rate models must be obtained from other sources. One source is the models developed for predicting IASCC crack growth rate (CGR) in BWR components. The other source is the models developed based on laboratory data that may include test results for both PWR and BWR representative conditions. These models were considered to be appropriate for the prediction of crack growth due to IASCC in PWR internals.
    - For weld locations subjected to fluence levels less than or equal to $5x10^{20}$ n/cm$^2$ (E>1MeV)(0.75 dpa), the maximum of the crack growth rate in 2017 ASME Section XI Appendix C-8520



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**W-ID: 4.2** **Core Barrel Assembly**

**Middle Axial Welds**

**or** the crack growth equation in ASME Code Case N-889 [52] will be used. The conditions for the use of Code Case N-889 identified in Regulatory Guide RG 1.147, Rev. 20 [67] must be met.

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

- Depending on the magnitude of cyclic loads, such as thermal transients, additional crack growth as a result of these loads may need to be considered.
- Acceptance criteria can be developed for the entire operating license of a given plant by using predicted end-of-license fluence values. However, re-inspection on the recommended intervals is required to confirm that the acceptance criteria are met. If there are changes to these fluence projections, such as in the event of a power uprate or change in core loading pattern, it would be necessary to confirm that the inputs selected based on fluence, such as SCC growth rate and fracture toughness, remain applicable or bounding until the end of the operating license.

**Failure**

| | |
|---|---|
| Failure Mechanism: | Cracking (SCC, IASCC) |
| Failure Effect: | Potential for stress redistribution within the core barrel, which could initiate or accelerate circumferential crack growth in the core barrel. |
| Failure Criteria: | Actively growing through-wall flaws may require mitigation.<br>An existing flaw is unacceptable if the flaw length projected at the next inspection cycle results in a potential loss of core support. |

**Methodology**

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



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**W-ID: 4.2** **Core Barrel Assembly**

**Middle Axial Welds**

Data Requirements:

1. Surface crack length as determined by visual inspection
    a. The 2013 edition of the ASME Code Section XI: IWA-3330(a) and Figure IWA-3330-1 provide requirements for the minimum allowable separation distance that will serve as the basis for guidance in this report. The requirement in Section XI is that the ligament between adjacent cracks must be greater than half of the thickness of the material. If this criterion is not met, the individual crack lengths and the length of the ligament between the cracks must be summed. This total length is then compared to the allowable length. Note that the 2007 and 2010 editions of ASME Code Section XI have the equivalent content to the 2013 edition and are also applicable.
    b. The flaw proximity rules in BWRVIP-158 [62] may be optionally applied in lieu of those defined in ASME Section XI: IWA-3330(a) and Figure IWA-3330-1 when appropriate.
2. Flaw Depth
    a. For one-sided visual inspections, the flaw is assumed to be through-wall.
    b. Supplemental examinations may be used to determine flaw depth, for a flaw-specific criterion, if needed.
3. Estimate fast neutron fluence at crack location.
4. Steady-state normal operating stresses to be used to calculate SCC crack growth rates.
    a. Hoop stresses based on normal mechanical loads are required for axial flaws.
    b. Stresses which have an insignificant net-through-wall value (average stress is near zero), such as weld residual stresses and thermal stresses due to local through-wall temperature gradients are considered to have minimal impact on the effective crack growth rates in through-wall flaws.
    c. Secondary weld residual and thermal stresses need to be considered in determination of axial and through-wall crack growth rates in partial through-wall flaws, whose dimensions would have to be determined with supplemental UT examinations.
5. Limiting externally applied transient stresses to be used to calculate allowable flaw lengths. Stresses arising from pressure, mechanical and thermal loads would be included in this calculation.
    a. More detailed load-deformation histories may be required for elastic-plastic or limit load calculations, if these calculations are necessary.



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**W-ID: 4.2** **Core Barrel Assembly**

**Middle Axial Welds**

Analysis: All analyses require an assumption of the SCC crack growth expected over the upcoming period of service. The methodology for analysis of a one-sided visual examination is based on an assumption of a through-wall flaw with weld residual and thermal stresses relieved. An appropriate crack growth rate model will be applied, which may be either K dependent or K independent. Fatigue crack growth has been assumed to be negligible. In order to apply the acceptance criteria to a period beyond one refueling cycle (not to exceed a 10-year re-inspection interval), additional action is required. The flaw evaluation will address the verification process of the predicted crack growth rate. Depending on the flaw size and knowledge of the plant conditions, a re-inspection at the next refueling outage may be required to provide data needed to operate beyond one cycle. The verification plan shall be included in the evaluation which is submitted to the regulator for their information. For detailed crack growth models used, see "Inputs and Assumptions" above.

Failure of the axial welds is assumed to occur when unstable axial crack growth is initiated from the analyzed flaw. Three options are outlined for determining the limiting allowable flaw length, based on neutron dose. Analysis methods are suggested for both pre-inspection or generic analysis (Suggested Pre-Inspection Analysis) and for flaws observed in-service (Suggested Flaw Specific Analysis), where more detailed characteristics of the flaw and its location are known. In all cases, a more detailed evaluation may be completed using a semi-elliptic surface flaw, but such an evaluation would require more detailed inspection by UT.



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**W-ID: 4.2** **Core Barrel Assembly**

**Middle Axial Welds**

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

Note:

1. Alternate fracture toughness values may be applied and justification for the alternate basis shall be submitted to the NRC for information within one year.
2. MRP-211 [66] provides the basis for the fracture toughness value.

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

Option 1: LEFM Analysis

a. Establish initial crack length (and depth if determined by supplemental examinations) based on inspection results.
b. The final crack dimensions are calculated by adding 10 years of crack growth under normal loading conditions.
c. Ensure that the $\Delta K$ resulting from flow-induced vibration (FIV) is below the threshold for fatigue crack growth.
d. For primary loading at normal and upset loading conditions, the final stress intensity factor (K) for the flaw size from (b) must be lower than the fracture toughness for the material by a factor of at least 2.77. For the evaluation of partial through wall flaws, the stress intensity factor due to secondary loading does not require an additional safety factor



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**Middle Axial Welds**

and may be directly added to the adjusted stress intensity factor for primary loading (i.e., $2.77 \cdot K_P + K_S < K_{IC}$).

e. For the governing primary loading at emergency or faulted loading conditions, the final stress intensity factor (K) for the flaw size from (b) must be lower than the fracture toughness for the material by a factor of at least 1.39. For the evaluation of partial through wall flaws, the stress intensity factor due to secondary loading does not require an additional safety factor and may be directly added to the adjusted stress intensity factor for primary loading (i.e., $1.39 \cdot K_P + K_S < K_{IC}$).

Option 2: EPFM Analysis (for neutron dose levels > 0.5 dpa but < 5 dpa)

a. Establish initial crack length (and depth if determined by supplemental examinations) based on inspection results.

b. The final crack dimensions are calculated by adding 10 years of crack growth under normal loading conditions.

c. Ensure that the ΔK resulting from flow-induced vibration (FIV) is below the threshold for fatigue crack growth.

d. Develop the (J/T) material curve from the material J-R curve.

e. The $J_{applied}$ curve must include a safety factor of 2.77 for primary loading at normal and upset conditions, and a factor of 1.39 for primary loading at emergency and faulted conditions. For the evaluation of partial through wall flaws, the J curve due to secondary loading does not require an additional safety factor and may be directly added to the adjusted J curve for primary loading (i.e., $[2.77 \text{ or } 1.39] \cdot J_p + J_s < J_{applied}$).

f. The intersection of the material and applied (J/T) curves indicates the instability point. The flaw size at instability is determined from the $J_{applied}$ versus flaw size curve.

g. The flaw size at instability must be larger than the flaw size from (b), for the flaw to be acceptable.

The intersection of the material and applied (J/T) curves indicates the instability point.

The load at instability is determined from the $J_{applied}$ versus load curve.

Option 3: Limit Load Analysis (For neutron fluence < $3 \times 10^{20}$ n/cm$^2$ at E>1 MeV or dose less than approximately 0.5 dpa)

a. Establish initial crack length (and depth if determined by supplemental examinations) based on inspection results.

b. The final crack dimensions are calculated by adding 10 years of crack growth under normal loading conditions.

c. Ensure that the ΔK resulting from flow-induced vibration (FIV) is below the threshold for fatigue crack growth.



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**Middle Axial Welds**

d. Determine the bending moment (M) that can be tolerated as a function of the postulated flaw length.

e. The applied moment, increased by a factor of 1.39 (for primary loading at emergency and faulted conditions) or 2.77 (for primary loading at normal and upset conditions) must be less than the limit moment (from step d) for the flaw length determined in (b), for the flaw to be acceptable. For the evaluation of partial through wall flaws, the moment due to secondary loading does not require an additional safety factor and may be directly added to the adjusted moment for primary loading (i.e., $[2.77 \text{ or } 1.39] \cdot M_p + M_s < M_{applied}$).

Acceptance Criteria: The middle axial welds continue to perform their functional requirements with the projected flaw lengths at the end of the inspection interval.

Approach: Limit load, EPFM, or LEFM analysis

Additional Action(s):

1) In order to apply the TR methodology for cracks detected via one-sided visual examinations, if supplementary examinations to determine crack depth were not performed, a follow up examination must be performed no later than the next RFO to confirm the assumed crack growth rates are conservative, unless a technical justification for a longer verification interval is submitted.

2) Note that the licensee shall submit the plant-specific crack growth and fracture mechanics analysis justifying the detected flaws, including a justification of the allowable operating period before re-inspection, within one year after any inspection that detects relevant conditions as defined in Table 5-3 of MRP-227.



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| W-ID: 4.3 | Core Barrel Assembly<br>Lower Axial Welds | | |
|---|---|---|---|
| Category: | Expansion | Applicability: | All plants |
| Degradation Effect: | See MRP-227 Table 4-6 | | |
| Primary Link: | See MRP-227 Table 4-6 | | |
| Function: | The lower axial welds contribute to the overall structural integrity of the core barrel. | | |

**Inspection**

| | |
|---|---|
| Method: | Visual examination. See MRP-227 Table 4-6. |
| Coverage: | See MRP-227 Table 4-6 |
| Observable Effect: | The specific relevant condition is a detectable crack-like surface indication. |

**Inputs and Assumptions**

There are several inputs and assumptions that are critical to the development of acceptance criteria for the RV internals lower axial welds. These items are stated below:

- The inspections identified in MRP-227 are intended to provide a sampling of potential locations of degradation. Under this approach, inspection of one side (surface) of the weld is assumed to provide an adequate sampling for monitoring of SCC/IASCC.
- The change in resistance to fracture of the RV lower axial welds can be correlated to the accumulated fluence at each weld location. Welds that are subject to low fluence are considered to have a high degree of resistance to fracture. Correspondingly, those welds subject to high fluence have lower resistance to fracture.
- It is assumed that the primary mode of crack growth to be expected in this region is SCC/IASCC.
- The prediction of crack growth is based on the stress intensity factor, K, calculated using linear elastic fracture mechanics. The rate of crack growth is dependent on the amount of neutron fluence that the weld is expected to accumulate over the licensed operating lifetime. Since there has been very limited experience of IASCC initiated cracks in this or comparable components or welds in operating PWRs to date, growth rate models must be obtained from other sources. One source is the models developed for predicting IASCC crack growth rate (CGR) in BWR components. The other source is the models developed based on laboratory data that may include test results for both PWR and BWR representative conditions. These models were considered to be appropriate for the prediction of crack growth due to IASCC in PWR internals.
    - For weld locations subjected to fluence levels less than or equal to $5x10^{20}$ n/cm$^2$ (E>1MeV)(0.75 dpa), the maximum of the crack growth rate in 2017 ASME Section XI Appendix C-8520



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**W-ID: 4.3** **Core Barrel Assembly**

**Lower Axial Welds**

**or** the crack growth equation in ASME Code Case N-889 [52] will be used. The conditions for the use of Code Case N-889 identified in Regulatory Guide RG 1.147, Rev. 20 [67] must be met.

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

- Depending on the magnitude of cyclic loads, such as thermal transients, additional crack growth as a result of these loads may need to be considered.
- Acceptance criteria can be developed for the entire operating license of a given plant by using predicted end-of-license fluence values. However, re-inspection on the recommended intervals is required to confirm that the acceptance criteria are met. If there are changes to these fluence projections, such as in the event of a power uprate or change in core loading pattern, it would be necessary to confirm that the inputs selected based on fluence, such as SCC growth rate and fracture toughness, remain applicable or bounding until the end of the operating license.

**Failure**

| | |
|---|---|
| Failure Mechanism: | Cracking (SCC, IASCC) |
| Failure Effect: | Potential for stress redistribution within the core barrel, which could initiate or accelerate circumferential crack growth in the core barrel. |
| Failure Criteria: | Actively growing through-wall flaws may require mitigation.<br>An existing flaw is unacceptable if the flaw length projected at the next inspection cycle results in a potential loss of core support. |

**Methodology**

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



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**Lower Axial Welds**

Data Requirements:

1. Surface crack length as determined by visual inspection
    a. The 2013 edition of the ASME Code Section XI: IWA-3330(a) and Figure IWA-3330-1 provide requirements for the minimum allowable separation distance that will serve as the basis for guidance in this report. The requirement in Section XI is that the ligament between adjacent cracks must be greater than half of the thickness of the material. If this criterion is not met, the individual crack lengths and the length of the ligament between the cracks must be summed. This total length is then compared to the allowable length. Note that the 2007 and 2010 editions of ASME Code Section XI have the equivalent content to the 2013 edition and are also applicable.
    b. The flaw proximity rules in BWRVIP-158 [62] may be optionally applied in lieu of those defined in ASME Section XI: IWA-3330(a) and Figure IWA-3330-1 when appropriate.
2. Flaw Depth
    a. For one-sided visual inspections, the flaw is assumed to be through-wall.
    b. Supplemental examinations may be used to determine flaw depth, for a flaw-specific criterion, if needed.
3. Estimate fast neutron fluence at crack location.
4. Steady-state normal operating stresses to be used to calculate SCC crack growth rates.
    a. Hoop stresses based on normal mechanical loads are required for axial flaws.
    b. Stresses which have an insignificant net-through-wall value (average stress is near zero), such as weld residual stresses and thermal stresses due to local through-wall temperature gradients are considered to have minimal impact on the effective crack growth rates in through-wall flaws.
    c. Secondary weld residual and thermal stresses need to be considered in determination of axial and through-wall crack growth rates in partial through-wall flaws, whose dimensions would have to be determined with supplemental UT examinations.
5. Limiting externally applied transient stresses to be used to calculate allowable flaw lengths. Stresses arising from pressure, mechanical and thermal loads would be included in this calculation.
    a. More detailed load-deformation histories may be required for elastic-plastic or limit load calculations, if these calculations are necessary.

Analysis: All analyses require an assumption of the SCC crack growth expected over the upcoming period of service. The methodology for analysis of a one-sided



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**Lower Axial Welds**

visual examination is based on an assumption of a through-wall flaw with weld residual and thermal stresses relieved. An appropriate crack growth rate model will be applied, which may be either K dependent or K independent. Fatigue crack growth has been assumed to be negligible. In order to apply the acceptance criteria to a period beyond one refueling cycle (not to exceed a 10-year re-inspection interval), additional action is required. The flaw evaluation will address the verification process of the predicted crack growth rate. Depending on the flaw size and knowledge of the plant conditions, a re-inspection at the next refueling outage may be required to provide data needed to operate beyond one cycle. The verification plan shall be included in the evaluation which is submitted to the regulator for their information. For detailed crack growth models used, see "Inputs and Assumptions" above.

Failure of the axial welds is assumed to occur when unstable axial crack growth is initiated from the analyzed flaw. Three options are outlined for determining the limiting allowable flaw length, based on neutron dose. Analysis methods are suggested for both pre-inspection or generic analysis (Suggested Pre-Inspection Analysis) and for flaws observed in-service (Suggested Flaw Specific Analysis), where more detailed characteristics of the flaw and its location are known. In all cases, a more detailed evaluation may be completed using a semi-elliptic surface flaw, but such an evaluation would require more detailed inspection by UT.



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**W-ID: 4.3** **Core Barrel Assembly**

**Lower Axial Welds**

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

Note:

1. Alternate fracture toughness values may be applied and justification for the alternate basis shall be submitted to the NRC for information within one year.
2. MRP-211 [66] provides the basis for the fracture toughness value.

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

Option 1: LEFM Analysis

a. Establish initial crack length (and depth if determined by supplemental examinations) based on inspection results.
b. The final crack dimensions are calculated by adding 10 years of crack growth under normal loading conditions.
c. Ensure that the $\Delta K$ resulting from flow-induced vibration (FIV) is below the threshold for fatigue crack growth.
d. For primary loading at normal and upset loading conditions, the final stress intensity factor (K) for the flaw size from (b) must be lower than the fracture toughness for the material by a factor of at least 2.77. For the evaluation of partial through wall flaws, the stress intensity factor due to secondary loading does not require an additional safety factor



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**Lower Axial Welds**

and may be directly added to the adjusted stress intensity factor for primary loading (i.e., $2.77 \cdot K_P + K_S < K_{IC}$).

e. For the governing primary loading at emergency or faulted loading conditions, the final stress intensity factor (K) for the flaw size from (b) must be lower than the fracture toughness for the material by a factor of at least 1.39. For the evaluation of partial through wall flaws, the stress intensity factor due to secondary loading does not require an additional safety factor and may be directly added to the adjusted stress intensity factor for primary loading (i.e., $1.39 \cdot K_P + K_S < K_{IC}$).

Option 2: EPFM Analysis (for neutron dose levels > 0.5 dpa but < 5 dpa)

a. Establish initial crack length (and depth if determined by supplemental examinations) based on inspection results.

b. The final crack dimensions are calculated by adding 10 years of crack growth under normal loading conditions.

c. Ensure that the ΔK resulting from flow-induced vibration (FIV) is below the threshold for fatigue crack growth.

d. Develop the (J/T) material curve from the material J-R curve.

e. The $J_{applied}$ curve must include a safety factor of 2.77 for primary loading at normal and upset conditions, and a factor of 1.39 for primary loading at emergency and faulted conditions. For the evaluation of partial through wall flaws, the J curve due to secondary loading does not require an additional safety factor and may be directly added to the adjusted J curve for primary loading (i.e., $[2.77 \text{ or } 1.39] \cdot J_p + J_s < J_{applied}$).

f. The intersection of the material and applied (J/T) curves indicates the instability point. The flaw size at instability is determined from the $J_{applied}$ versus flaw size curve.

g. The flaw size at instability must be larger than the flaw size from (b), for the flaw to be acceptable.

The intersection of the material and applied (J/T) curves indicates the instability point. The load at instability is determined from the $J_{applied}$ versus load curve.

Option 3: Limit Load Analysis (For neutron fluence < $3x10^{20}$ n/cm$^2$ at E>1 MeV or dose less than approximately 0.5 dpa)

a. Establish initial crack length (and depth if determined by supplemental examinations) based on inspection results.

b. The final crack dimensions are calculated by adding 10 years of crack growth under normal loading conditions.

c. Ensure that the ΔK resulting from flow-induced vibration (FIV) is below the threshold for fatigue crack growth.



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**W-ID: 4.3** **Core Barrel Assembly**

**Lower Axial Welds**

d. Determine the bending moment (M) that can be tolerated as a function of the postulated flaw length.

e. The applied moment, increased by a factor of 1.39 (for primary loading at emergency and faulted conditions) or 2.77 (for primary loading at normal and upset conditions) must be less than the limit moment (from step d) for the flaw length determined in (b), for the flaw to be acceptable. For the evaluation of partial through wall flaws, the moment due to secondary loading does not require an additional safety factor and may be directly added to the adjusted moment for primary loading (i.e., $[2.77 \text{ or } 1.39] \cdot M_p + M_s < M_{applied}$).

Acceptance Criteria: The lower axial welds continue to perform their functional requirements with the projected flaw lengths at the end of the inspection interval.

Approach: Limit load, EPFM, or LEFM analysis

Additional Action(s):

1) In order to apply the TR methodology for cracks detected via one-sided visual examinations, if supplementary examinations to determine crack depth were not performed, a follow up examination must be performed no later than the next RFO to confirm the assumed crack growth rates are conservative, unless a technical justification for a longer verification interval is submitted.

2) Note that the licensee shall submit the plant-specific crack growth and fracture mechanics analysis justifying the detected flaws, including a justification of the allowable operating period before re-inspection, within one year after any inspection x1cmthat detects relevant conditions as defined in Table 5-3 of MRP-227.



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|---|---|
| **W-ID: 4.4** | **Lower Support Assembly**<br>**Lower Support Column Bodies (Both Cast and Non-Cast)** |
| Category: | Expansion Applicability: All plants |
| Degradation Effect: | See MRP-227 Table 4-6 |
| Primary Link: | See MRP-227 Table 4-6 |
| Function: | The lower support columns provide the structural link between the lower core plate that supports the fuel assemblies and the lower support forging/casting. The supports are required to keep the lower core plate from deforming during operation. |
| **Inspection** | |
| Method: | Visual examination. See MRP-227 Table 4-6. |
| Coverage: | See MRP-227 Table 4-6 |
| Observable Effect: | Fracture<br>Potential for core tilt<br>Control rod insertion problems |
| **Failure** | |
| Failure Mechanism: | The upper sections of the core supports may experience neutron fluences above the threshold for IASCC. Although stresses in columns are primarily compressive, bending stresses or the design of the attachment may produce localized regions of tensile stress. |
| Failure Effect: | Displacement of lower core plate |
| Failure Criteria: | Must maintain sufficient number of intact support columns to assure dimensional stability of lower core plate. |
| **Methodology** | |
| Goal: | Establish acceptable pattern of lower support column bodies. |
| Data Requirements: | Applicable normal/upset and emergency/faulted loads on the lower support structure.<br>Design information including geometry, materials, and method of fabrication for lower core plate, columns, and barrel.<br>Fluence accumulated by the lower support column bodies. |
| Analysis: | 1. Establish functional requirements for lower support column bodies:<br>a. Normal/upset requirement: During the enveloping case of normal and upset service conditions operation, the system of support columns should resist core support plate deformation due to mechanical or thermal loading.<br>b. Emergency/faulted requirement: During emergency and faulted service conditions loadings, the core support system must maintain |



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**W-ID: 4.4** **Lower Support Assembly**

**Lower Support Column Bodies (Both Cast and Non-Cast)**

structural integrity, ensure core coolable geometry, and ensure control rod insertability.

c. Applicable stress limits shall be met in the core support columns to confirm the validity of the analytical approach, consistent with the design basis.

2. Support column analysis assumptions.
   a. Assume any column with a crack in the main body to have failed.
   b. Assume any column with a crack in the weld to result in failure of the attachment.
3. Structural model of lower support structure.

   Create a structural model of the lower support structure that includes support columns and the lower core plate. The model should be capable of simulating postulated failure in individual columns, thus producing results to confirm that functional requirements are met.
4. The structural model must be run for functional requirements A, B, and C.

Acceptance Criteria: The observed pattern of failed columns must meet the defined acceptable column pattern for both normal/upset and emergency/faulted conditions.

There is currently no basis for estimating column failure rates. Therefore, application of the acceptance criteria requires sufficient margin to provide allowance for additional flawed columns that is equal to the observed number of flawed columns.

N = Number of core support columns

$N_{req}$ = Number of core support columns required to meet acceptable pattern

Margin = $N-N_{req}$

In addition to meeting the acceptable patterns, the maximum allowable number of flawed columns is defined as follows:

$N_f$ = Number of observed flawed columns

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

Approach: Pattern analysis specific to the lower support structure design and loads.

Note that a PWROG functionality analysis is complete.

Additional Action(s): Note that the licensee shall submit the plant-specific analysis for the acceptable distribution of intact components within one year after any inspection that detects the relevant conditions as defined in Table 5-3 of MRP-227.



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|---|---|
| **W-ID: 5** | **Baffle-former Assembly**<br>**Baffle-edge Bolts** |
| Category: | Primary Applicability: All plants with baffle-edge bolts |
| Degradation Effect: | See MRP-227 Table 4-3 |
| Expansion Link: | See MRP-227 Table 4-3 |
| Function: | The baffle-edge bolts provide the baffle plate-to-baffle plate attachment along the seam between plates. The edge bolts prevent gaps between plates that can result in baffle jetting damage to peripheral fuel assemblies.<br>Studies have demonstrated that baffle edge bolts are not required to maintain the structural integrity of the baffle. |
| **Inspection** | |
| Method: | Visual examination. See MRP-227 Table 4-3. |
| Coverage: | See MRP-227 Table 4-3 |
| Observable Effect: | Failure of bolt or locking device as listed under inspection. |
| **Failure** | |
| Failure Mechanism: | Analysis has shown that differential thermal expansion and swelling can cause plastic deformation of edge bolts. These bolts are in high radiation locations and there is a significant potential failure due to IASCC.<br>Failure modes considered should include:<br>• Broken or missing locking devices<br>• Protruding bolt heads<br>• Missing bolts or bolt heads |
| Failure Effect: | In plants with downward coolant flow in the region between the baffle-former assembly and the core barrel, failure may contribute to baffle jetting.<br>Primary concerns are loose parts generation and interference with fuel. |
| Failure Criteria: | All bolts and locking devices should be in place and undamaged. |
| **Methodology** | |
| Goal: | A plant-specific plan should be developed for evaluating and mitigating the potential relevant conditions. The evaluation should consider any previously reported observations. |
| Data Requirements: | FMEA results |
| Analysis: | Observation of relevant conditions in the baffle edge bolts or locking bars would require a FMEA on the observed condition. It is recommended that the utility prepare for examination by conducting a failure modes and effects analysis (FMEA) for a full range of potential relevant observations prior to inspection. |



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**W-ID: 5** **Baffle-former Assembly**

**Baffle-edge Bolts**

Confirmation of aging as the driver for any relevant observations is critical. Degradation at a limited number of isolated edge bolt locations would not be taken as an indication of aging.

Basic steps for conducting the FMEA would include:

1. Defining the system to be analyzed with clearly identified system boundaries
2. Breaking the system down into complete and mutually exclusive elements, including functional and physical aspects
3. Identifying failure modes, effects, and criticality
   - Determine how the component item can fail
   - Evaluate what will happen to the system and its environment if the component fails in each of the identified failure modes
   - Assign criticality metric values to assess susceptibility or failure likelihood and severity of consequences
4. Include a broad range of expert panel expertise to participate in the process

At a minimum, a FMEA for the baffle-former assembly and baffle-edge bolts should consider the following types of indications that could be observed by the visual VT-3 inspection:

Plates
- Markings/gouges
- Separation (gaps)
- Warping
- Cracks

Locking devices
- Broken
- Damaged
- Missing
- Improperly installed

Bolts
- Protruding head
- Missing
- Broken
- Improperly installed

Consequences resulting from these observed conditions are:

1. Baffle jetting
2. Interference with Fuel assemblies
3. Fuel damage
4. Flow blockage



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| | |
|---|---|
| **W-ID: 5** | **Baffle-former Assembly**<br>**Baffle-edge Bolts** |
| | One example of a consequence that might be observed as a result of a degraded condition is a protruding bolt or a warped or separated plate which impedes the loading and unloading of fuel. Another example would be fuel degradation resulting from interaction between a fuel assembly and a degraded bolt, lock bar, or plate. Replacing the affected bolt would be an example solution for resolving a bolting degradation issue. These are examples of consequences that are typically considered in the baffle-former assembly and baffle-edge bolt FMEA. |
| Acceptance Criteria: | Determined by FMEA |
| Approach: | FMEA should address plant-specific practices and priorities. Some generic work possible to outline issues and options to be addressed in FMEA. |
| Additional Action(s): | Note that the licensee shall submit the plant-specific or generic FMEA analysis and confirmation of applicability of generic FMEA analysis within one year after any inspection that detects relevant conditions as defined in Table 5-3 of MRP-227. |



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



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**W-ID: 6** **Baffle-former Assembly**

**Baffle-former Bolts**

2. The degradation is categorized as either "Typical" or "Accelerated" based upon plant design and the inspection results. The percentage of bolts with indications is based upon the fraction of the total quantity. This categorization considers the likelihood of additional bolt failures due to IASCC over a typical 10-year re-inspection interval. The degradation category for the observed and assumed failed bolts shall be determined using the following table:

| Plant Design | Inspection Results(2) | Degradation Category |
|---|---|---|
| Downflow | Less than 3% indications with no clustering(1) | Typical |
| | 3% or greater indications or clustering(1) | Accelerated |
| Upflow and Converted Upflow | Less than 5% indications with no clustering(1) | Typical |
| | 5% or greater indications or clustering(1) | Accelerated |

Notes:

1. Clustering is defined in MRP 2017-009 as three or more adjacent defective baffle-former bolts (BFBs) in the vertical or lateral directions on the same baffle plate, or more than 40% defective BFBs on the same baffle plate. Sampling statistics may be applied on a plant-specific basis to credit untestable bolts as structural members when classifying clustering.
2. These criteria are considered applicable to subsequent inspections until an appropriate degradation rate threshold is established through operating experience.

3. Additional margin to account for potential degradation through the next re-inspection interval must be provided. Plants that exhibit "Accelerated" degradation must apply the Probabilistic Model, but those with "Typical" degradation may apply either the Probabilistic Model or the Margin Ratio method as defined in the Acceptance Criteria.

Acceptance Criteria:

1. The analysis requirements as outlined above are applied to ensure structural integrity of the bolting pattern.
2. The degradation category is established consistent with industry guidance.
3. The evaluation must include margin by applying one of the following methods:



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**W-ID: 6** **Baffle-former Assembly**

**Baffle-former Bolts**

Probabilistic Model (Typical or Accelerated Degradation)

The evaluation applies the following process steps

a. Develop an empirical and/or mechanistic bolt degradation model from plant operating experience, plant-specific design information and test data. The model is calibrated to plant-specific bolt inspection results, or otherwise applies bolt inspection results from similar plants after accounting for relevant differences as appropriate.

b. Determine patterns of predicted failed bolts at the end of the re-inspection interval based upon the likelihood of failure. Justification shall be provided that an appropriate degradation rate has been applied. The evaluation is allowed to credit bolt replacements, or plant improvements such as upflow conversion.

c. A probabilistic reliability analysis is used to demonstrate reasonable assurance that the pattern of failed bolts will be acceptable at the end of the re-inspection interval. These techniques must consider both the total proportion of failed bolts and the spatial distribution, or pattern, of bolt failures. When predicting the total proportion of failed bolts, it is acceptable to consider results associated with the 95th fractile. For the potential spatial distribution of bolt failures, an evaluation showing acceptability of at least 95% of the predicted patterns at the end of the re-inspection interval can be used. It is anticipated that this method will be used to demonstrate a 10 year re-inspection interval. Re-inspection intervals beyond 10 years may be justifiable in certain circumstances (e.g., acceptable bolting pattern relies on only replacement bolts and the plant is in, or has been converted to, an upflow configuration) based upon submittal of a deviation to MRP-227 requirements.

EPRI has provided general guidelines for the minimum suggested content for probabilistic fracture mechanics submittals in letter BWRVIP 2019-016 [63]. At a high level based on this document, the minimum content for the bolt probabilistic reliability analysis can be summarized as:

- Description of model and underlying principals
- Required plant-specific or bounding inputs (bolt stress, dose, and inspection results)
  - Basis for uncertainties for each input
  - Treatment of input uncertainty in the evaluation
- Summary of benchmarking and convergence studies demonstrating the validity of the model and model results

It is recognized that due to the proprietary nature of some inputs or models used in bolt probabilistic evaluations, portions of this content may not be included directly in the submittal. Per BWRVIP 2019-016, the submitter should have a plan for making the supporting materials available if needed for NRC review of the submittal.



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**W-ID: 6** **Baffle-former Assembly**

**Baffle-former Bolts**

Margin Ratio (Typical Degradation only)

Less than 50% of the initial margin may be consumed. Therefore, the quantity of bolt failures upon re-inspection must be less than or equal to the number of bolt failures that has occurred to date.

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

Where

N = total number of baffle-former bolts

$N_{req}$ = number of baffle-former bolts in acceptable pattern

$N_f$ = number of failed bolts

The margin (M) is

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

This degradation model may be applied at subsequent re-inspections, up to a 10-year re-inspection interval, when the "Typical" degradation category remains applicable.

Approach: Generic ABPAs that were previously performed should be confirmed to be applicable on a plant-specific basis.

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

Additional Action(s): 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.



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| **W-ID: 6.1** | **Core Barrel Assembly**<br>**Barrel-former Bolts** |
| Category: | Expansion Applicability: All plants |
| Degradation Effect: | See MRP-227 Table 4-6 |
| Primary Link: | See MRP-227 Table 4-6 and Interim Guidance MRP 2018-002 [48] |
| Function: | Maintain structural integrity of baffle-former-barrel structure. |
| **Inspection** | |
| Method: | Volumetric examination. See MRP-227 Table 4-6 |
| Coverage: | See MRP-227 Table 4-6 |
| Observable Effect: | Indication of flaw from UT examination |
| **Failure** | |
| Failure Mechanism: | Cracking<br>Loss of bolt pre-load due to irradiation induced stress relaxation may exacerbate fatigue issue in aging plants |
| Failure Effect: | Potential for flow induced vibration due to loss of bolting constraint.<br>Loss of structural stability |
| Failure Criteria: | Require an acceptable bolting pattern. |
| **Methodology** | |
| Goal: | Must demonstrate an acceptable bolting pattern. |
| Data Requirements: | Loads/displacements<br>Bolting patterns<br>Baffle-former-barrel design<br>Fast neutron (dpa) distribution in baffle-barrel region (if needed)<br>Projected bolt failure rate |
| Analysis: | Procedures for establishing acceptable bolting patterns for the barrel-to-former bolts have been established in WCAP-15029-P-A [13] / WCAP-15030-NP-A [14] as supplemented by PWROG-18034-P-A [64] and PWROG-18034-NP-A [65]. This methodology has been reviewed and accepted by the NRC in a Safety Evaluation issued in 1998 (TAC No. MA1152). |
| Acceptance Criteria: | 1. Observed pattern of unfailed bolts meets pre-defined acceptance criteria or is analyzed in real time.<br>2. Less than 50% of initial margin consumed.<br>$N_f < (N - N_{req})/2$ (See below)<br>The observed pattern of failed bolts must be shown to meet the conditions for being an acceptable bolt pattern per [13] / [14] as supplemented by [64] / [65] and have a reasonable margin to protect against additional failures during the inspection interval. The margin is defined in terms of the number of intact bolts |



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

**Barrel-former Bolts**

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

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

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

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

Approach: Generic work completed in previous PWROG program.

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

Additional Action(s): None



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| | |
|---|---|
| **W-ID: 6.2** | **Lower Support Assembly**<br>**Lower Support Column Bolts** |
| Category: | Expansion Applicability: All plants |
| Degradation Effect: | See MRP-227 Table 4-6 |
| Primary Link: | See MRP-227 Table 4-6 |
| Function: | The lower support column bolts attach the support columns to the lower core plate. Although the bolts do not directly support the weight of the core, they help maintain the flatness and integrity of the lower support plate. |
| **Inspection** | |
| Method: | Volumetric examination. See MRP-227 Table 4-6 |
| Coverage: | See MRP-227 Table 4-6 |
| Observable Effect: | Indication of flaw from UT examination |
| **Failure** | |
| Failure Mechanism: | Cracking |
| Failure Effect: | Displacement of lower core plate |
| Failure Criteria: | Determine acceptable number of support columns required to maintain structural integrity. |
| **Methodology** | |
| Goal: | Conservatively assume that failure of a bolt results in loss of attachment between the support column and the lower core plate. Establish functional requirements for core support columns.<br>1. During normal operation system of support columns should resist core plate deformation due to mechanical or thermal loading. Core plate requirements for "flatness" and fuel assembly alignment.<br>2. During limiting accident transient system must maintain structural integrity. |
| Data Requirements: | Loads on lower core plate<br>Displacement tolerances on lower core plate<br>Fluence accumulated by the lower support column bolts<br>Constitutive model for stainless steel properties as a function of irradiation |
| Analysis: | 1. Establish functional requirements for core support columns.<br>a. During normal operation system of support columns should resist core plate deformation due to mechanical or thermal loading. Core plate requirements for "flatness" and fuel assembly alignment.<br>b. During limiting accident transient system must maintain structural integrity. |



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**W-ID: 6.2** **Lower Support Assembly**

**Lower Support Column Bolts**

2. Structural model of lower support structure.
   FEA model of lower support structure that includes support columns and lower core plate. Model should be capable of removing individual column or breaking attachment to lower core plate. Would require multiple iterations to establish "acceptable patterns" of core support columns and support column bolts.
3. Structural model must be run for functional requirements A and B.
4. Determine margin for additional failures.
   a. Assume number of failures in next 10 years is equal to number observed to date.

      N = # of Support Columns

      $N_f$ = # of Observed Flawed Columns

      $N_{req}$ = # of columns in relevant acceptable pattern

      Margin = $N - N_{req}$

| | |
|---|---|
| Acceptance Criteria: | $N_f < (N - N_{req})/2$ (See below) |
| Approach: | Generic program to share first-of-a-kind effort.<br>Pilot analysis of lower support structure to identify critical issues.<br>Expect final acceptance based on plant-specific consideration. |
| Additional Action(s): | Note that the licensee shall submit the plant-specific analysis for the acceptable distribution of intact components within one year after any inspection that triggers the expansion criteria as defined in Table 5-3 of MRP-227. |



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| W-ID: 7 | Baffle-former Assembly<br>Assembly | | |
|---|---|---|---|
| Category: | Primary | Applicability: | All plants |
| Degradation Effect: | See MRP-227 Table 4-3 | | |
| Expansion Link: | See MRP-227 Table 4-3 | | |
| Function: | Provide support, guidance, and protection for the reactor core<br>Provide a passageway for the distribution of the reactor coolant flow to the reactor core<br>Provide gamma and neutron shielding for the reactor vessel | | |

**Inspection**

| | |
|---|---|
| Method: | Visual examination. See MRP-227 Table 4-3. |
| Coverage: | See MRP-227 Table 4-3 |
| Observable Effect: | Degradation of general condition as described above |

**Failure**

| | |
|---|---|
| Failure Mechanism: | Void swelling, IASCC |
| Failure Effect: | 1. Interference with fuel assemblies<br>2. Obstruction of coolant flow<br>3. Loose parts generation<br>4. Distortion/misalignment of core<br>5. Local temperature peaks<br>6. Degradation of control rod insertability<br>7. Baffle jetting |
| Failure Criteria: | No relevant observations |

**Methodology**

| | |
|---|---|
| Goal: | A plant-specific plan should be developed for evaluating and mitigating the potential relevant conditions. The evaluation should consider any previously reported observations. |
| Data Requirements: | 1. Baseline data on previous visual examinations of baffle-former assembly<br>2. Loose parts monitoring data |
| Analysis: | Prepare for examination by conducting a failure modes and effects analysis (FMEA) to identify full range of potential relevant observations prior to inspection. Basic steps for conducting the FMEA would include:<br>1. Defining the system to be analyzed with clearly identified system boundaries<br>2. Breaking the system down into complete and mutually exclusive elements, including functional and physical aspects |



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**W-ID: 7** **Baffle-former Assembly**

**Assembly**

3. Identifying failure modes, effects, and criticality
   - Determine how the component item can fail
   - Evaluate what will happen to the system and its environment if the component fails in each of the identified failure modes
   - Assign criticality metric values to assess susceptibility or failure likelihood and severity of consequences
4. Include a broad range of expert panel expertise to participate in the process

Failure mechanisms considered should include:

- Broken or missing locking devices
- Protruding bolt heads
- Missing bolts or bolt heads
- Distortion or displacement of baffle plates
- Gross cracking of baffle plates
- Gaps at plate joints
- Interaction with fuel assemblies
- Historical record

At a minimum, a FMEA for the baffle-former assembly and baffle-former bolts should consider the following types of indications that could be observed by the visual VT-3 inspection:

Plates
- Markings/gouges
- Separation (gaps)
- Warping
- Cracks

Locking device
- Broken
- Damaged
- Missing
- Improperly installed

Bolts
- Protruding head
- Missing
- Broken
- Improperly installed



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| | |
|---|---|
| **W-ID: 7** | **Baffle-former Assembly**<br>**Assembly** |
| | Consequences resulting from these observed conditions are:<br><br>1. Baffle jetting<br>2. Interference with Fuel assemblies<br>3. Fuel damage<br>4. Flow blockage<br><br>One example of a consequence that might be observed as a result of a degraded condition is a protruding bolt or a warped or separated plate which impedes the loading and unloading of fuel. Another example would be fuel degradation resulting from interaction between a fuel assembly and a degraded bolt, lock bar, or plate. Replacing the affected bolt would be an example solution for resolving a bolting degradation issue. These are examples of consequences that are typically considered in the baffle-former assembly and baffle-edge bolt FMEA. |
| Acceptance Criteria: | Determined by FMEA |
| Approach: | FMEA should address plant-specific practices and priorities. Some generic work possible to outline issues and options to be addressed in FMEA. |
| Additional Action(s): | Note that the licensee shall submit the plant-specific or generic FMEA analysis and confirmation of applicability of generic FMEA analysis within one year after any inspection that detects relevant conditions as defined in Table 5-3 of MRP-227. |



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| W-ID: 8 | Alignment and Interfacing Components<br>Internals Hold Down Spring | | |
|---|---|---|---|
| Category: | Primary | Applicability: | All plants with 304 stainless steel hold down springs |
| Degradation Effect: | See MRP-227 Table 4-3 | | |
| Expansion Link: | See MRP-227 Table 4-3 | | |
| Function: | Provide hold down forces for core internals.<br>Retain internals in proper alignment to the core. | | |

| **Inspection** | |
|---|---|
| Method: | See MRP-227 Table 4-3 |
| Coverage: | See MRP-227 Table 4-3 |
| Observable Effect: | Reduced height of core hold down spring. Repeated measurements should indicate progressive reduction in height from cycle-to-cycle. Wear surfaces may also exhibit evidence of galling. |

| **Failure** | |
|---|---|
| Failure Mechanism: | Stress relaxation |
| Failure Effect: | Loss of hold down forces may lead to vibration and wear in lower internals |
| Failure Criteria: | Failure to maintain hold down force through next inspection cycle |

| **Methodology** | |
|---|---|
| Goal: | Remaining spring force must meet requirements for core hold down forces. |
| Data Requirements: | • Historical information on spring height (project rate of relaxation)<br>• Effective spring constant<br>• Necessary hold down force (plant-specific)<br>• Current spring height<br>• Trending of prior readings, if any |
| Analysis: | Need to construct creep-stress relaxation model to define bounding (high relaxation) behavior.<br>• Material properties (stiffness, creep)<br>• History (transients, creep)<br>• Geometry<br>• Force profile |



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| | |
|---|---|
| **W-ID: 8** | **Alignment and Interfacing Components**<br>**Internals Hold Down Spring** |
| Acceptance Criteria: | Relaxation of hold down spring must be above bounding prediction.<br>Projection to end of inspection interval must assure that hold down force is maintained through next inspection interval. |
| Approach: | Value determined by plant-specific design requirements. |
| Additional Action(s): | None |

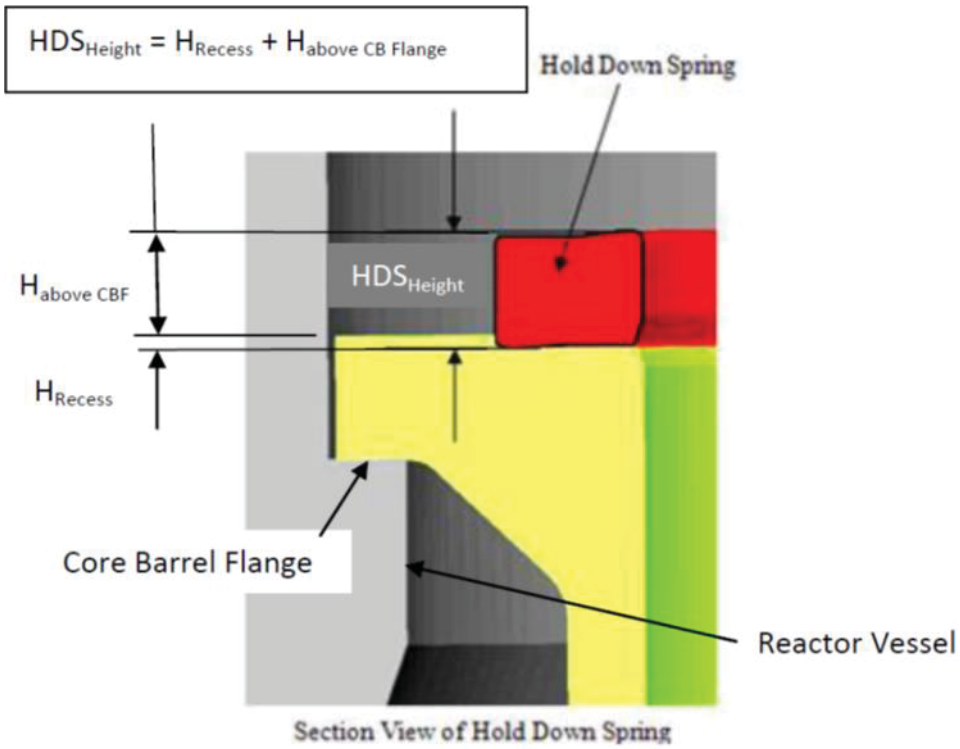


**Figure W-ID: 8-1 Hold Down Spring Height Measurement**



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| W-ID: 9 | **Thermal Shield Assembly**<br>**Thermal Shield Flexures** |
|---|---|
| Category: | Primary Applicability: All plants |
| Degradation Effect: | See MRP-227 Table 4-3 |
| Expansion Link: | See MRP-227 Table 4-3 |
| Function: | The flexure is the lower structural support for the thermal shield. Flexures hold the thermal shield concentric to the core. The flexure design allows for differential thermal expansion between the core barrel and the thermal shield. |

**Inspection**

| | |
|---|---|
| Method: | Visual examination. See MRP-227 Table 4-3. |
| Coverage: | See MRP-227 Table 4-3. |
| Observable Effect: | Crack, displacement, fracture, or component separation.<br>Failure along weld at base of flexure or failure of weld attachment to thermal shield. |

**Inputs and Assumptions**

Assume:

1. Any thermal shield flexure with an observed flaw has failed.
2. No credit for “bumpers” and other redundant structures.

**Failure**

| | |
|---|---|
| Failure Mechanism: | Large deflections of the flexure due to thermal cycling may lead to fatigue failures. |
| Failure Effect: | Failure of flexures contributes to vibration of the thermal shield. Failure can also result in flow blockage, wear, and damage to specimen guides. |
| Failure Criteria: | Number of unfailed thermal shield flexures must be sufficient to retain structural functionality of the entire thermal shield assembly. |

**Methodology**

| | |
|---|---|
| Goal: | Determine the number and location of thermal shield flexures that must remain intact to retain structural functionality of the entire thermal shield assembly. |
| Data Requirements: | • Load: thermal, deadweight, pressure, seismic, FIV, LOCA (if applicable)<br>• Design-specific Geometry<br>• History of transient cycles<br>• Materials of construction |



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**W-ID: 9** **Thermal Shield Assembly**

**Thermal Shield Flexures**

Analysis:

Perform structural assessment to determine the minimum number of flexures required to retain structural integrity.

The dynamic response of the thermal shield should be established.

Procedure:

1. Construct and benchmark an FEA model of the undamaged core barrel and thermal shield assembly for the plant.
2. Perform a modal analysis to determine the frequencies of at least the first twenty modes (as a minimum the cantilever beam mode and shell modes n = 2, n = 3, and n = 4 should be included).
3. Remove one flexure from the model and determine the effect on the thermal shield mode frequencies and the effect on the structural integrity of the remaining thermal shield supports.
4. If the frequency for a given mode changes, evaluate high cycle fatigue to demonstrate that the endurance limit is not exceeded in the remaining thermal shield supports. Normal operating stresses, including stresses from flow-induced vibration, must be considered.
5. Assess low cycle fatigue in the remaining supports, the top (blocks) and bottom (flexure) supports, including flexures, blocks, pins, and bolts for the interval before the next inspection cycle. Stresses resulting from normal and upset load conditions, including cyclic thermal loads, as well as seismic and LOCA conditions (LOCA loads may not be required for some plants), must be considered.
6. Remove additional flexures from the model and repeat steps 3 through 5 if the results of the frequency and fatigue assessment are acceptable for the failure of one flexure.

Acceptance Criteria:

Failure of a thermal shield flexure is acceptable if it can be demonstrated that the remaining thermal shield supports are structurally acceptable when the flexure is removed from the model. The resulting loads and stresses in the remaining flexures, bolts, and pins must be evaluated to determine the expected time to failure. If the dynamic response of the thermal shield is changed by removal of one flexure, then resulting loads and stresses in the remaining flexures, bolts, and pins must be evaluated to determine the expected time to failure. This will determine the required maximum permissible interval for re-inspection.

Any observation of a failed thermal shield flexure should lead to enhanced vigilance for fatigue and vibration monitoring systems. Enhanced vigilance could involve several approaches, such as, more frequent inspection during outages and neutron noise monitoring. Re-inspection should be performed in the first outage following discovery of a failed flexure and every outage thereafter. If neutron noise monitoring is selected and visual examination during the first outage following discovery finds no additional flaws, further visual examination is not required until the next 10-year inspection as long as neutron noise monitoring is conducted every 30 days and no change in the vibration frequencies is detected.



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**W-ID: 9** **Thermal Shield Assembly**

**Thermal Shield Flexures**

| | |
|---|---|
| Approach: | Plant-specific analysis. Could be adapted from a generic dynamic model. |
| Additional Action(s): | Note that plant-specific analysis of acceptable dynamic response of the remaining flexures for this component must be submitted to the NRC to determine if review and approval is needed, within one year after any inspection that detects relevant conditions as defined in the "Examination Acceptance Criteria" column in Table 5-3 of MRP-227. |



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# APPENDIX F
# FLOWCHARTS FOR WESTINGHOUSE-DESIGN PRIMARY AND EXPANSION COMPONENT ITEM INSPECTIONS

**Westinghouse Primary and Expansion Components**

W-ID: 1 Control Rod Guide Tube Assembly – Guide Plates (Cards)
W-ID: 2 Control Rod Guide Tube Assembly – Lower Flange Welds

- W-ID: 2.1 Control Rod Guide Tube Assembly – Remaining CRGT Lower Flange Welds
- W-ID: 2.2 Bottom-Mounted Instrumentation (BMI) System – BMI Column Bodies

W-ID: 3 Core Barrel Assembly – Upper Flange Weld

- W-ID: 3.1 Core Barrel Assembly – Upper Girth Weld
- W-ID: 3.2 Core Barrel Assembly – Upper Axial Weld
- W-ID: 3.3 Core Barrel Assembly – Lower Flange Weld
- W-ID: 3.4 Lower Internals Assembly – Lower Support Forging or Casting

W-ID: 4 Core Barrel Assembly – Lower Girth Weld

- W-ID: 4.1 Upper Internals Assembly – Upper Core Plate
- W-ID: 4.2 Core Barrel Assembly – Middle Axial Welds
- W-ID: 4.3 Core Barrel Assembly – Lower Axial Welds
- W-ID: 4.4 Lower Support Assembly – Lower Support Columns (Both Cast and Non-Cast)

W-ID: 5 Baffle-former Assembly – Baffle-Edge Bolts
W-ID: 6 Baffle-former Assembly – Baffle-Former Bolts

- W-ID: 6.1 Core Barrel Assembly – Barrel-Former Bolts
- W-ID: 6.2 Lower Support Assembly – Lower Support Column Bolts

W-ID: 7 Baffle-Former Assembly – Assembly

W-ID: 8 Alignment and Interfacing Components – Internals Hold Down Spring

W-ID: 9 Thermal Shield Assembly – Thermal Shield Flexures



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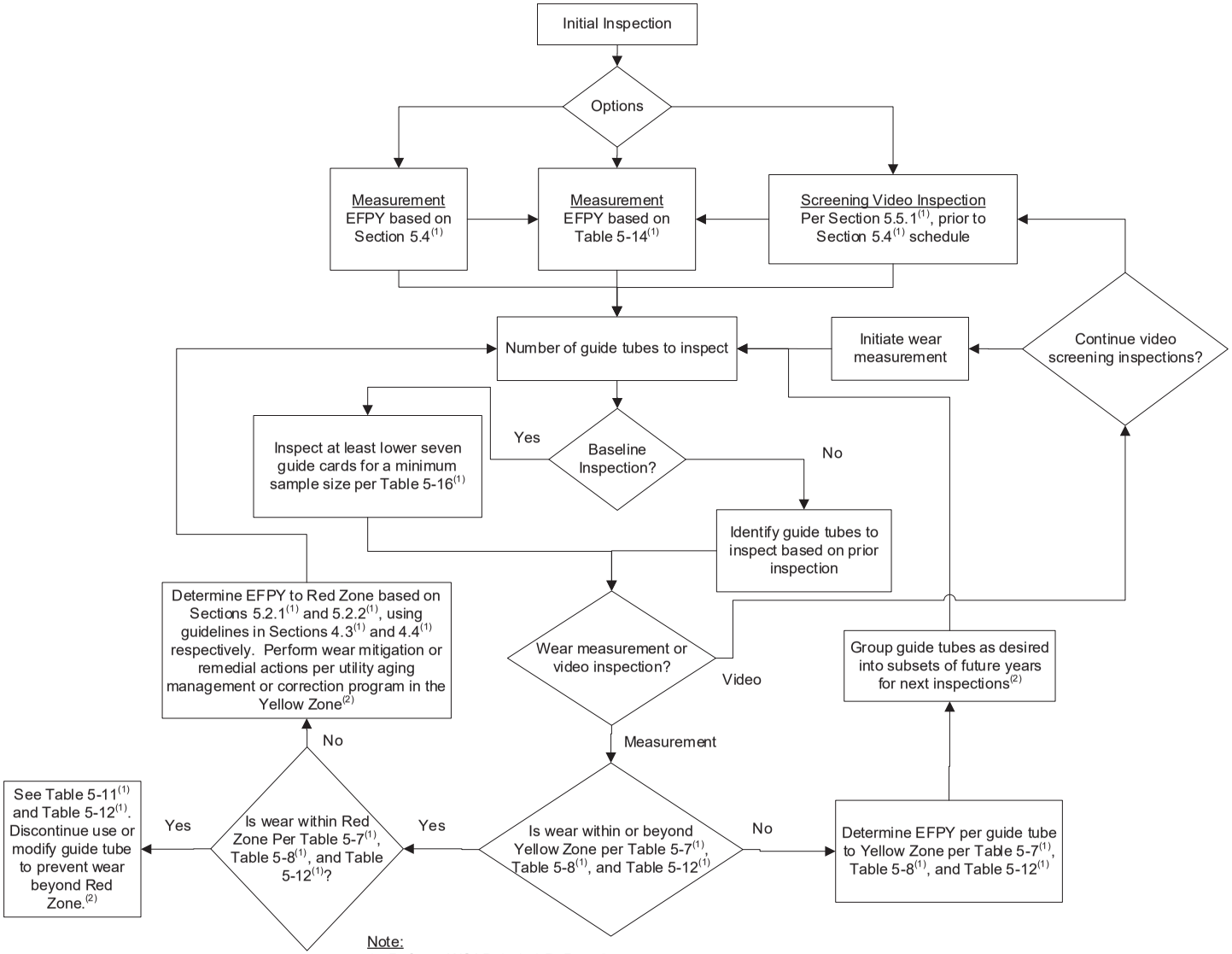
W-ID:1
Control Rod Guide Tube Assembly - Guide Plates (Cards)




Note:
1. Refer to WCAP-17451-P, Rev. 2
2. Perform any subsequent "Additional Actions" to be completed by the licensee when necessary



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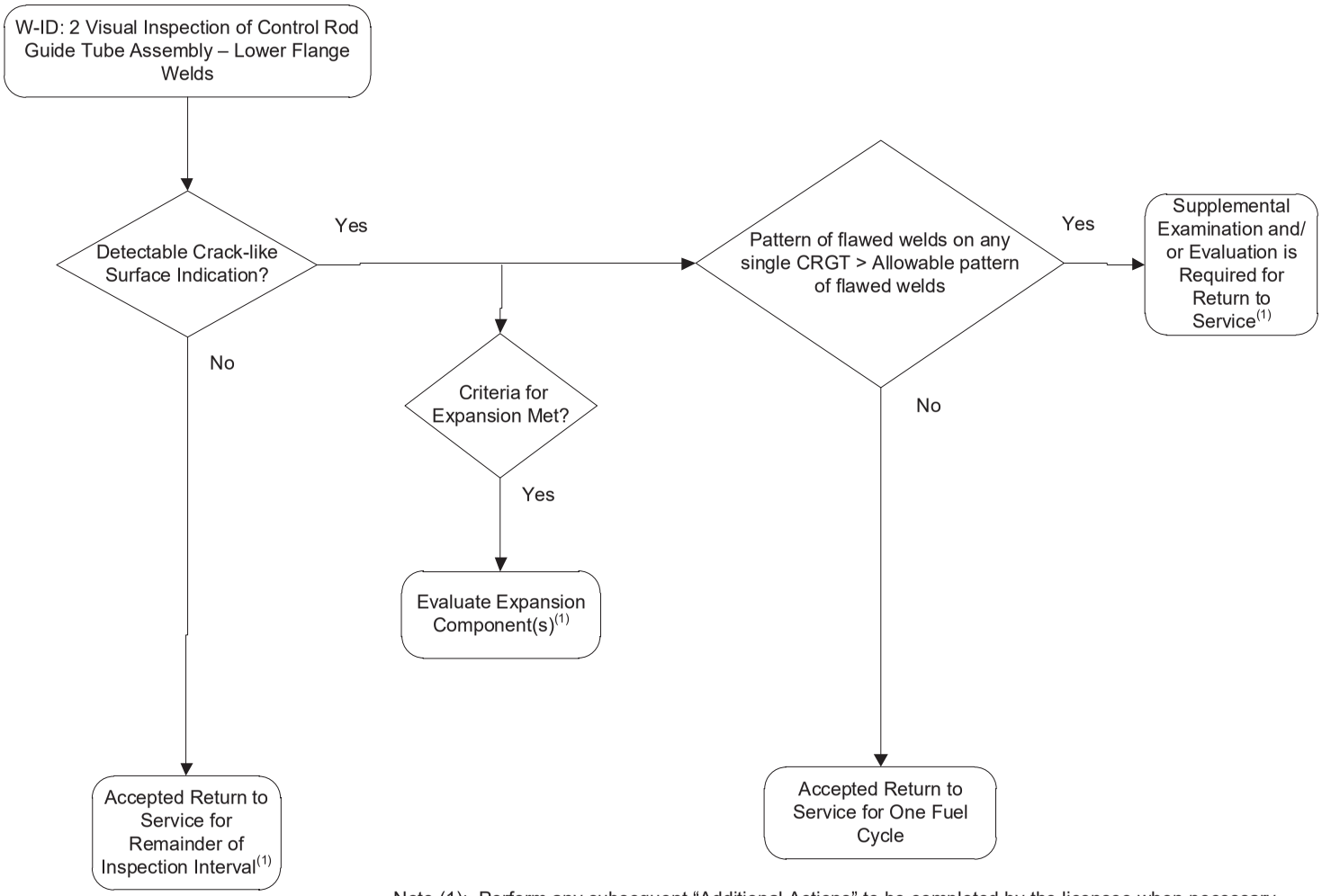
## W-ID: 2
## Control Rod Guide Tube Assembly – Lower Flange Welds

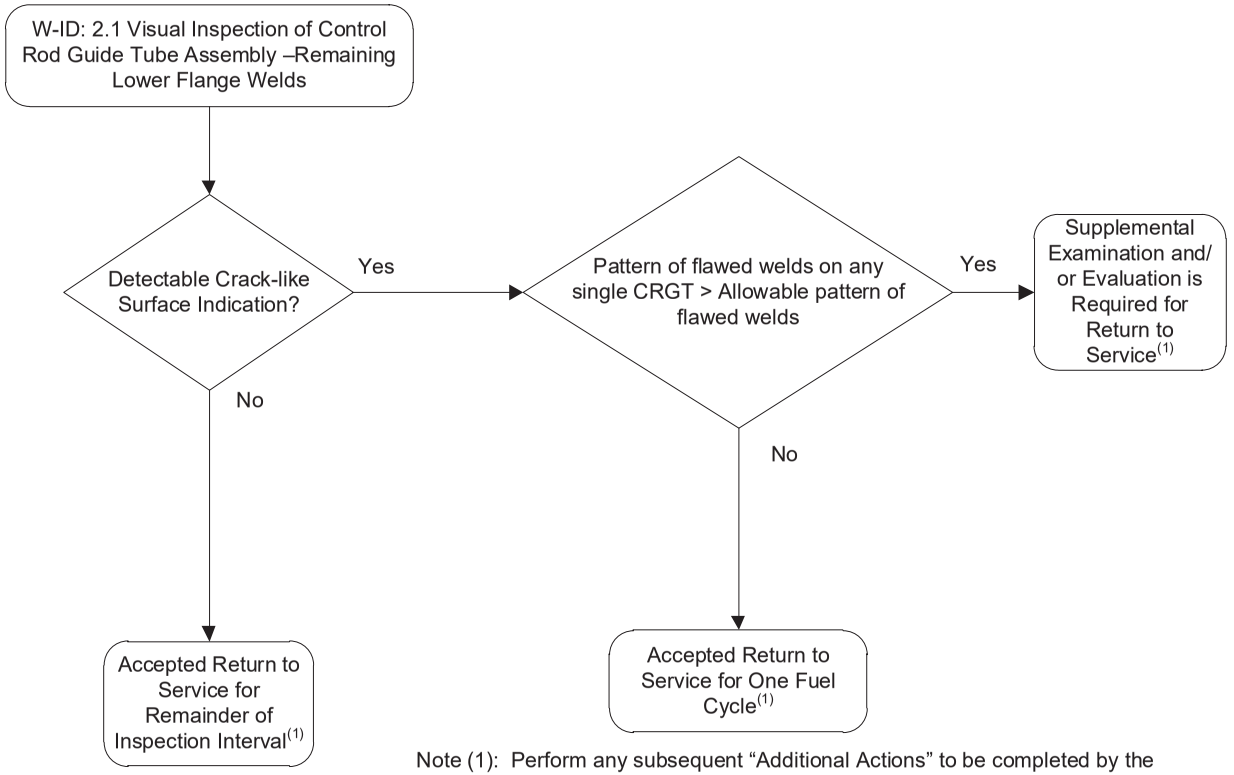



Note (1): Perform any subsequent “Additional Actions” to be completed by the licensee when necessary



*** This record was final approved on 08/31/2023 13:45:20. (This statement was added by the PRIME system upon its validation)

W-ID: 2.1
Control Rod Guide Tube Assembly – CRGT Remaining Lower Flange Welds




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



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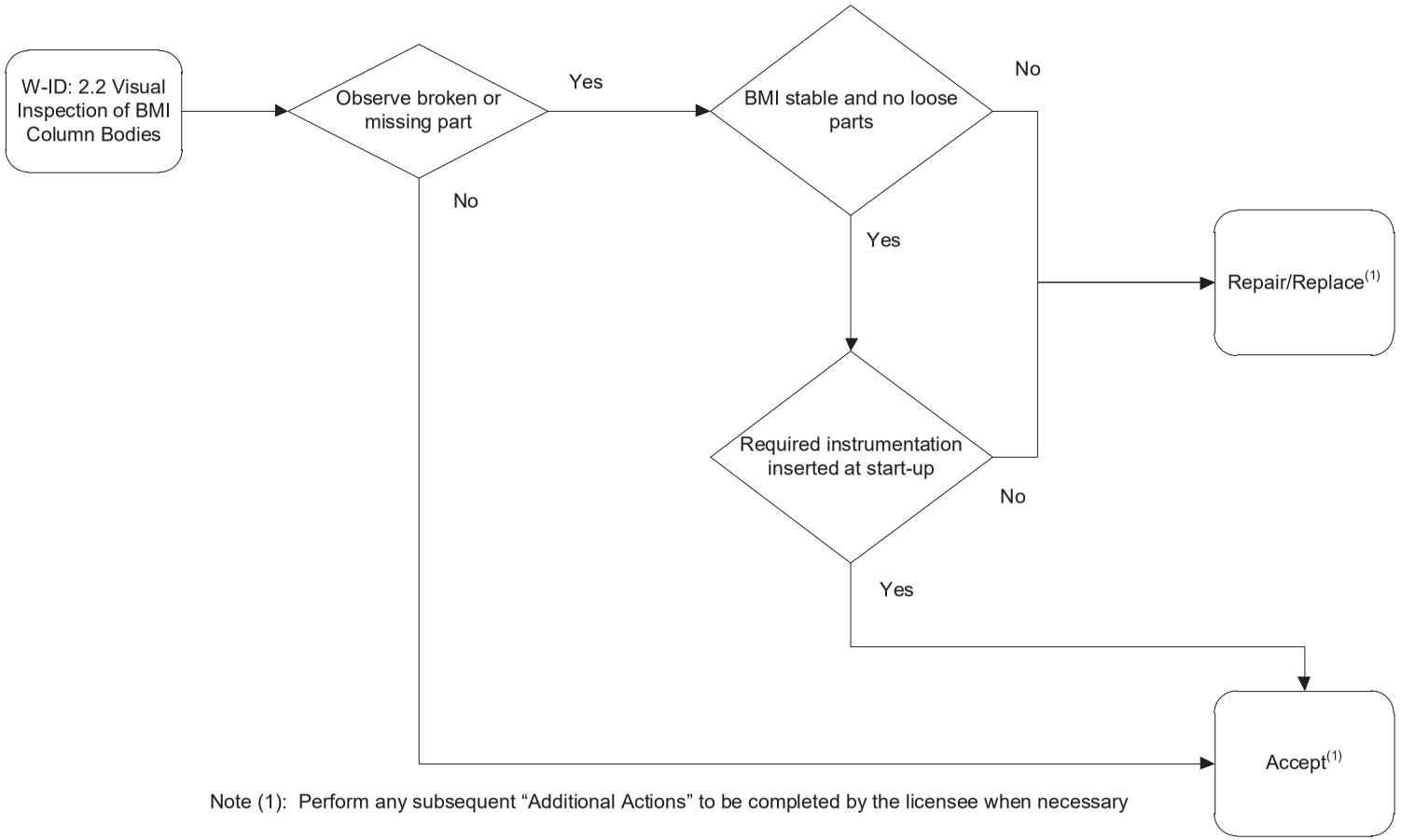
W-ID:2.2
Bottom Mounted Instrumentation (BMI) System - BMI Column Bodies

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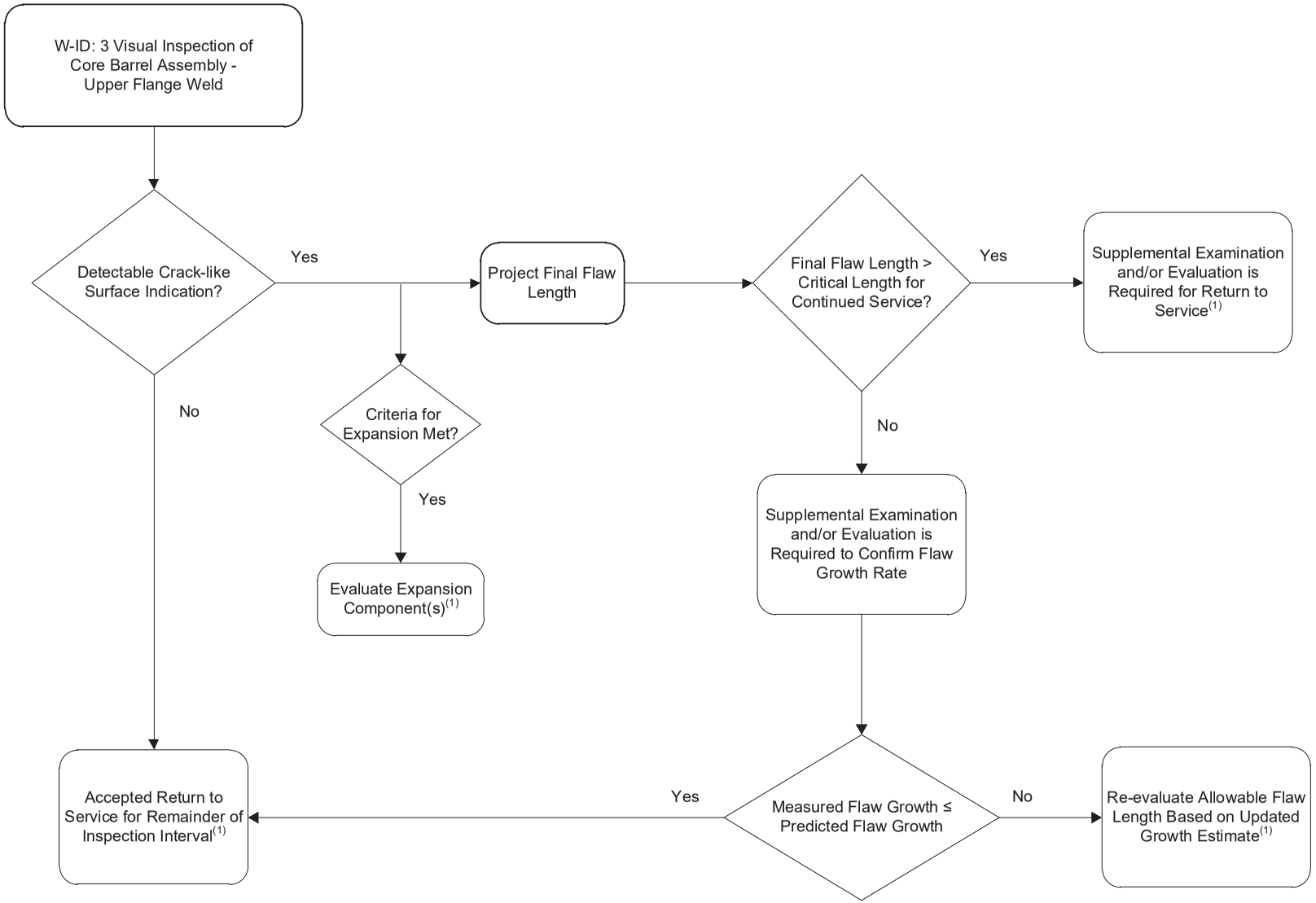
W-ID:3
Core Barrel Assembly - Upper Flange Weld

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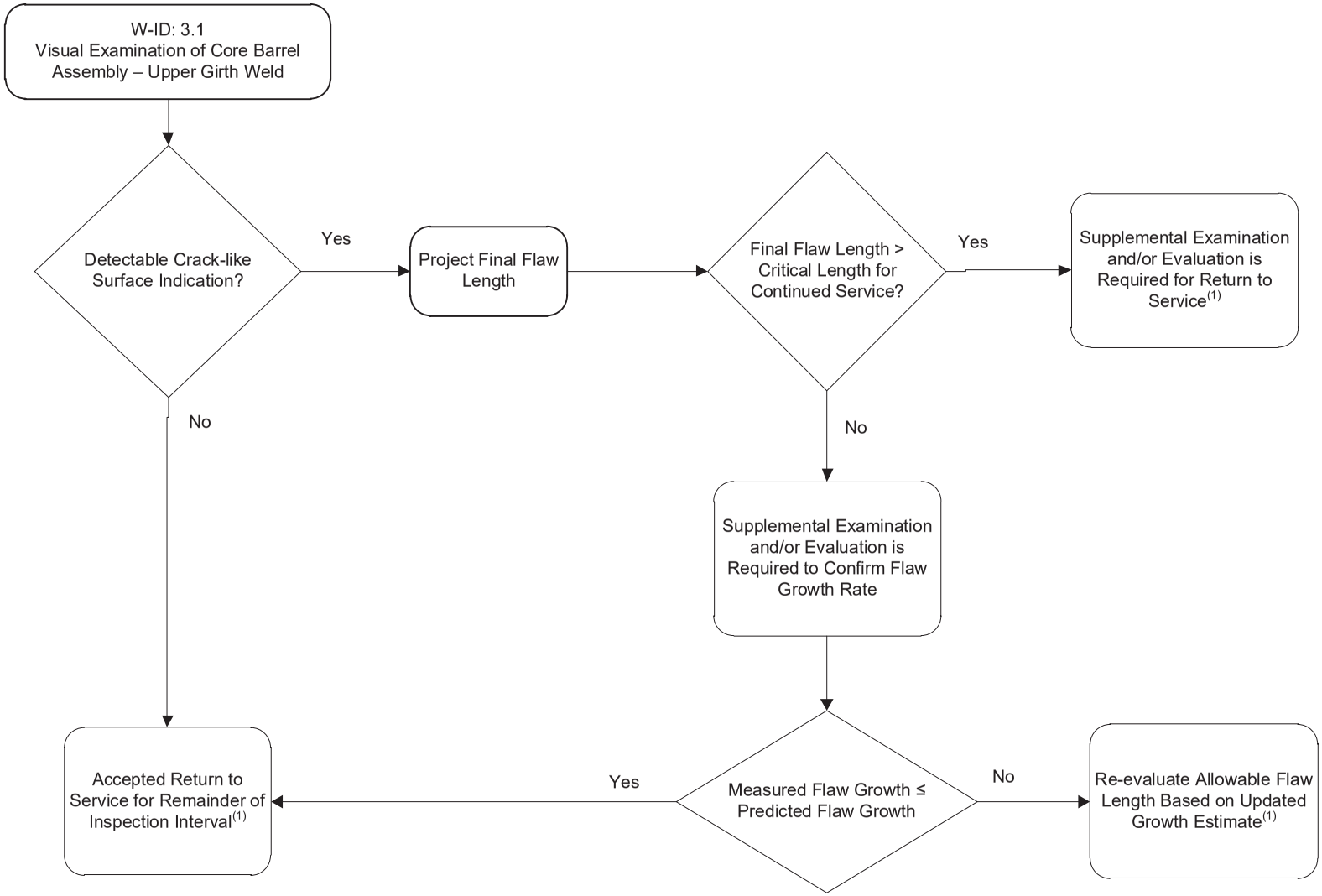
## W-ID: 3.1
## Core Barrel Assembly – Upper Girth Weld

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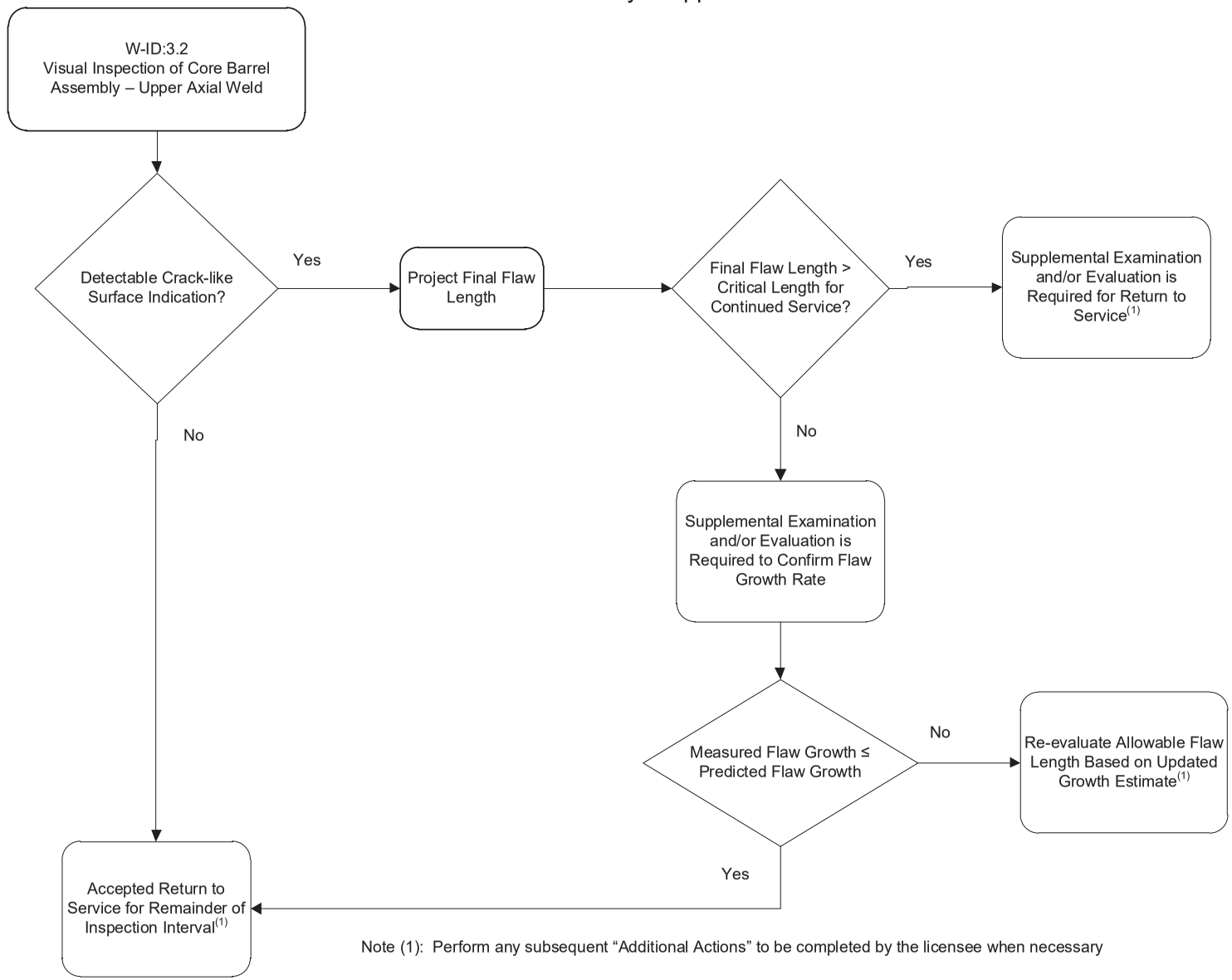
W-ID: 3.2
Core Barrel Assembly – Upper Axial Weld




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



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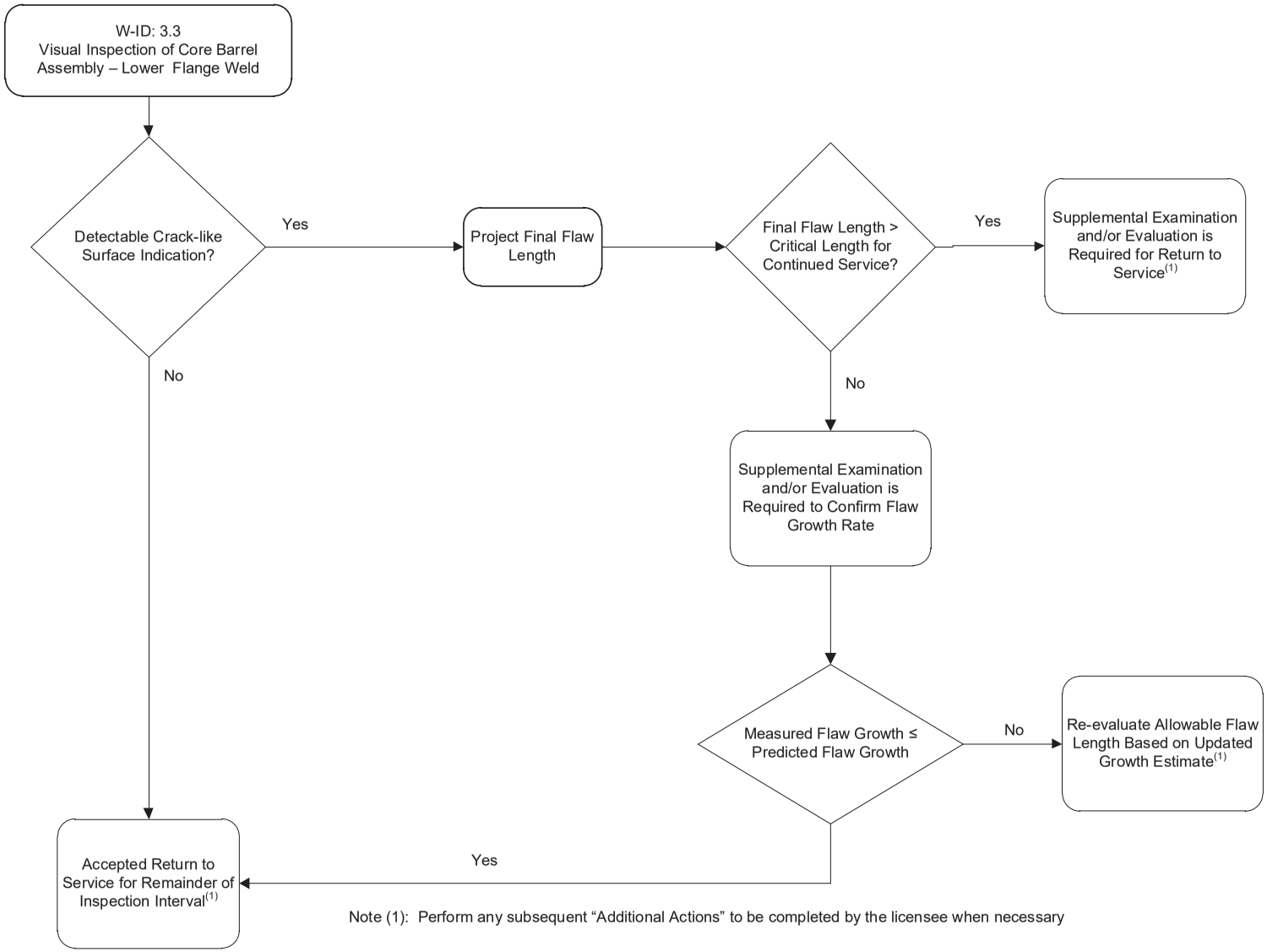
W-ID: 3.3
Core Barrel Assembly – Lower Flange Weld




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



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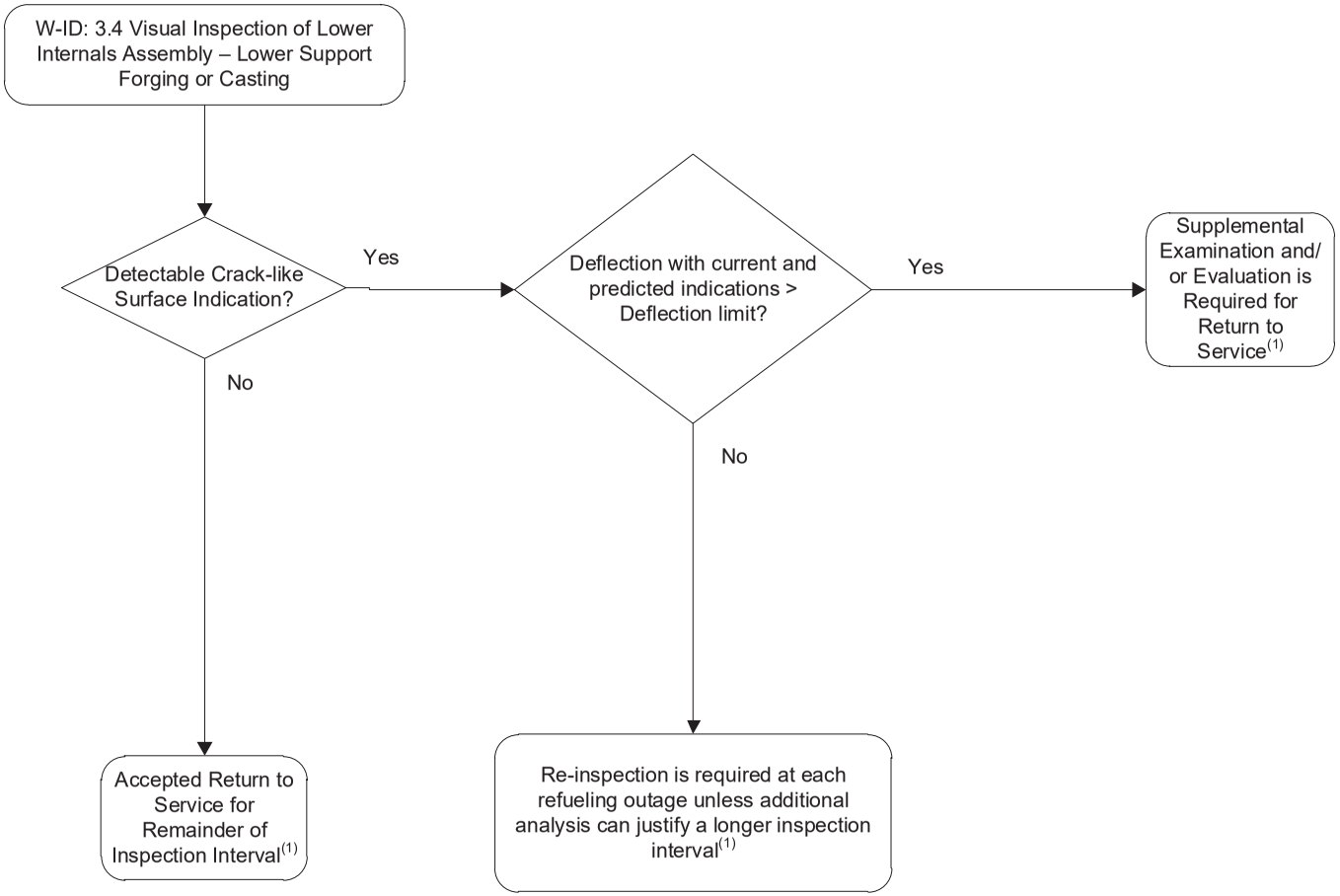
## W-ID: 3.4
## Lower Internals Assembly – Lower Support Forging or Casting




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



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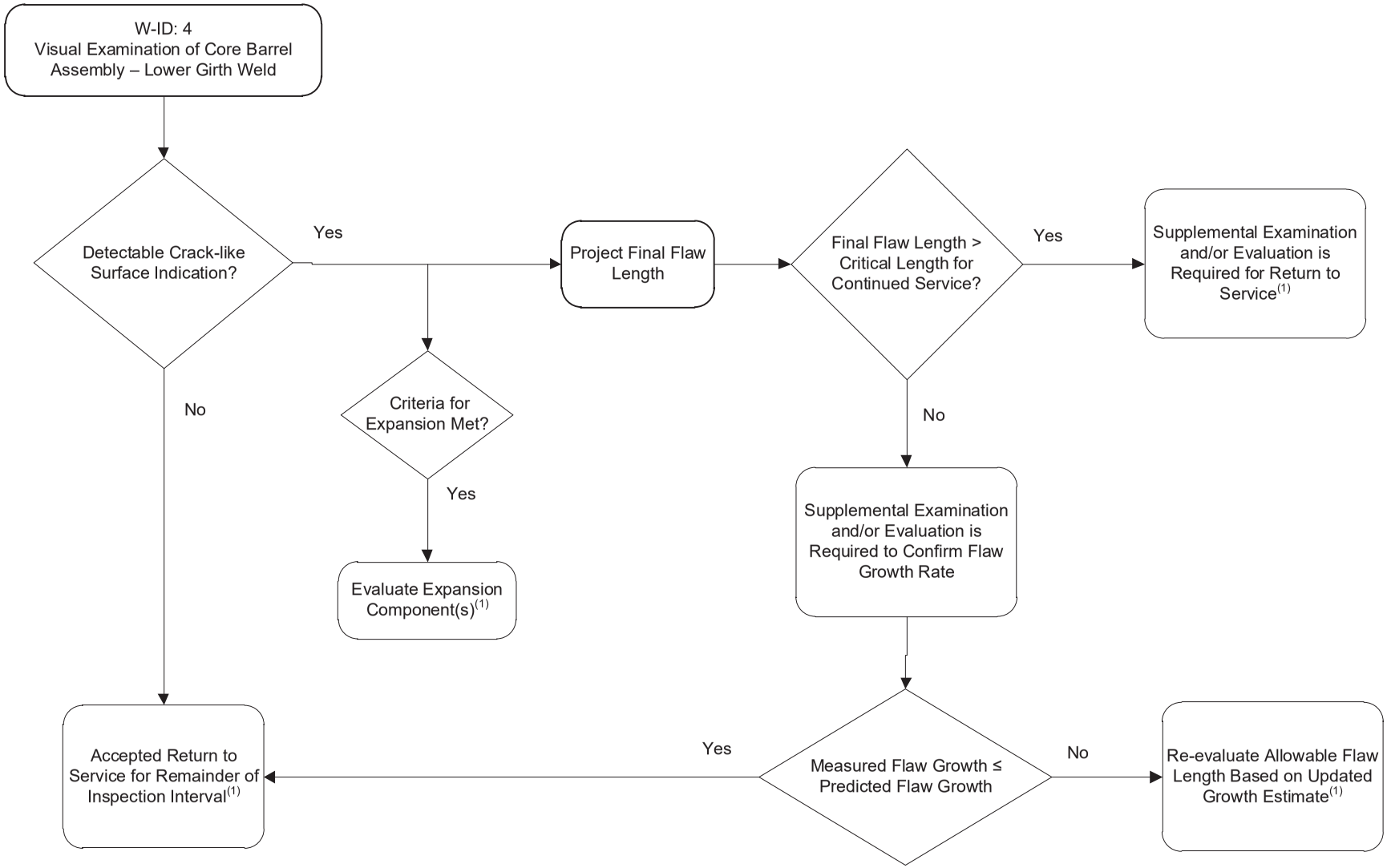
# W-ID: 4
# Core Barrel Assembly – Lower Girth Weld

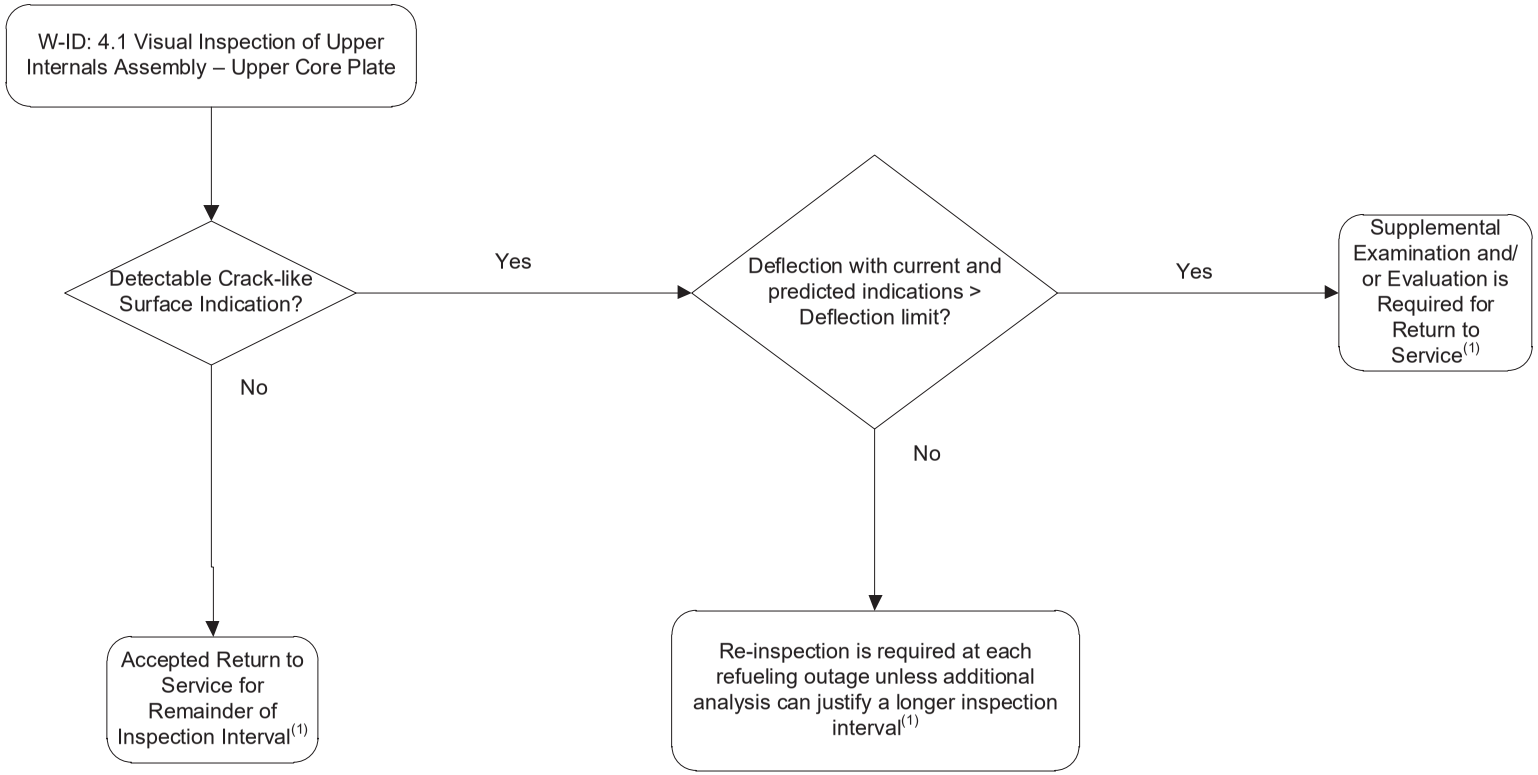



Note (1): Perform any subsequent “Additional Actions” to be completed by the licensee when necessary



*** This record was final approved on 08/31/2023 13:45:20. (This statement was added by the PRIME system upon its validation)

W-ID: 4.1
Upper Internals Assembly – Upper Core Plate

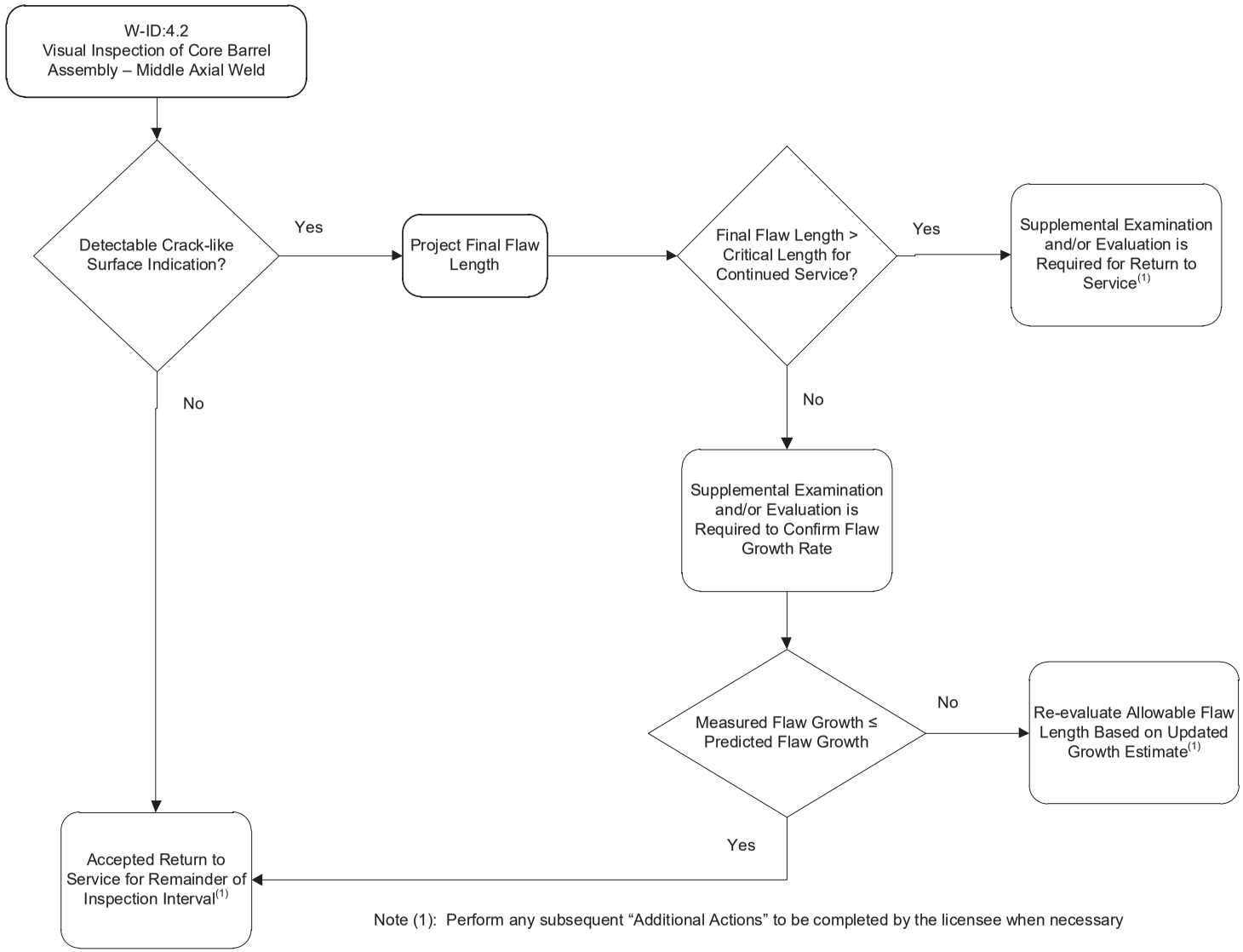



Note (1): Perform any subsequent “Additional Actions” to be completed by the licensee when necessary



*** This record was final approved on 08/31/2023 13:45:20. (This statement was added by the PRIME system upon its validation)

W-ID: 4.2
Core Barrel Assembly – Middle Axial Welds

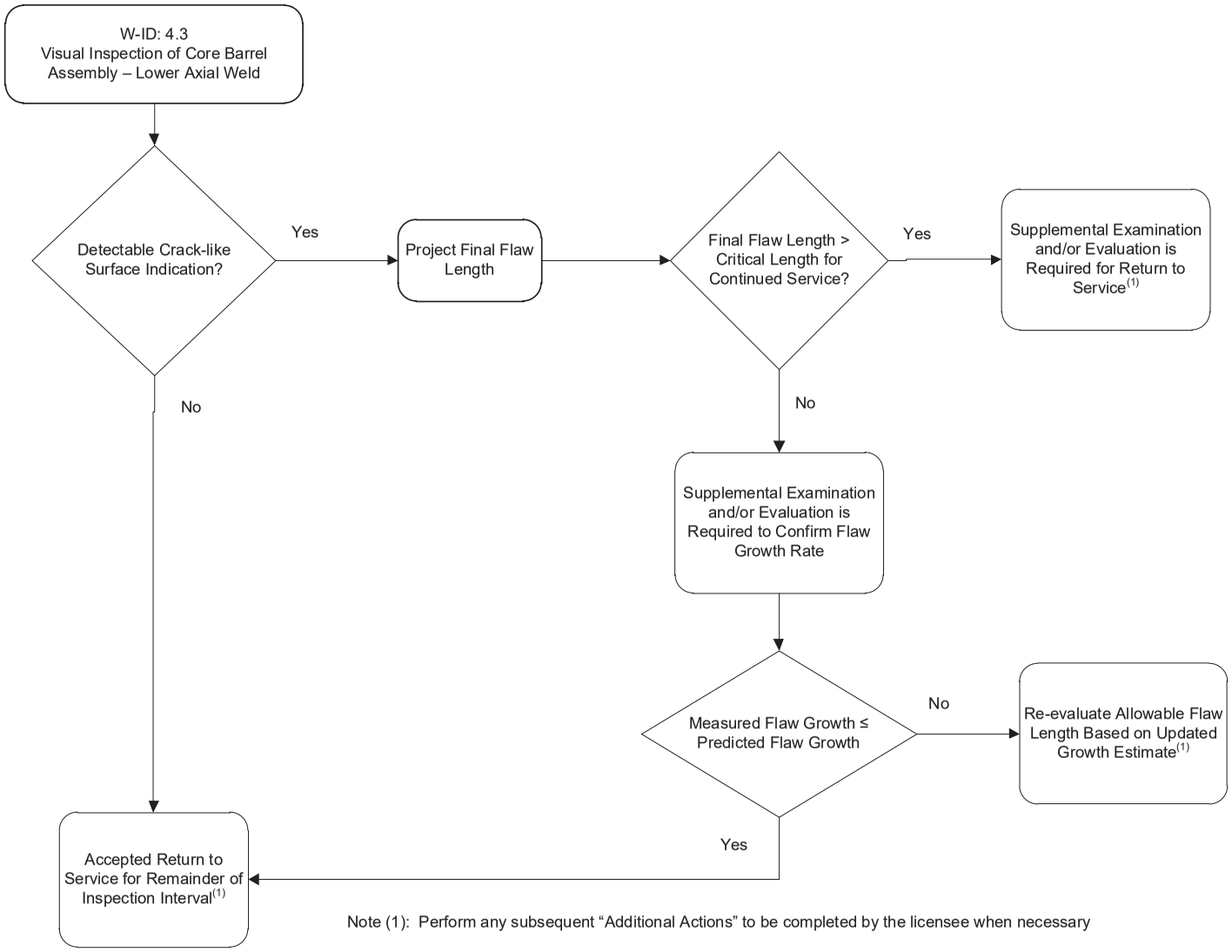



Note (1): Perform any subsequent "Additional Actions" to be completed by the licensee when necessary



*** This record was final approved on 08/31/2023 13:45:20. (This statement was added by the PRIME system upon its validation)

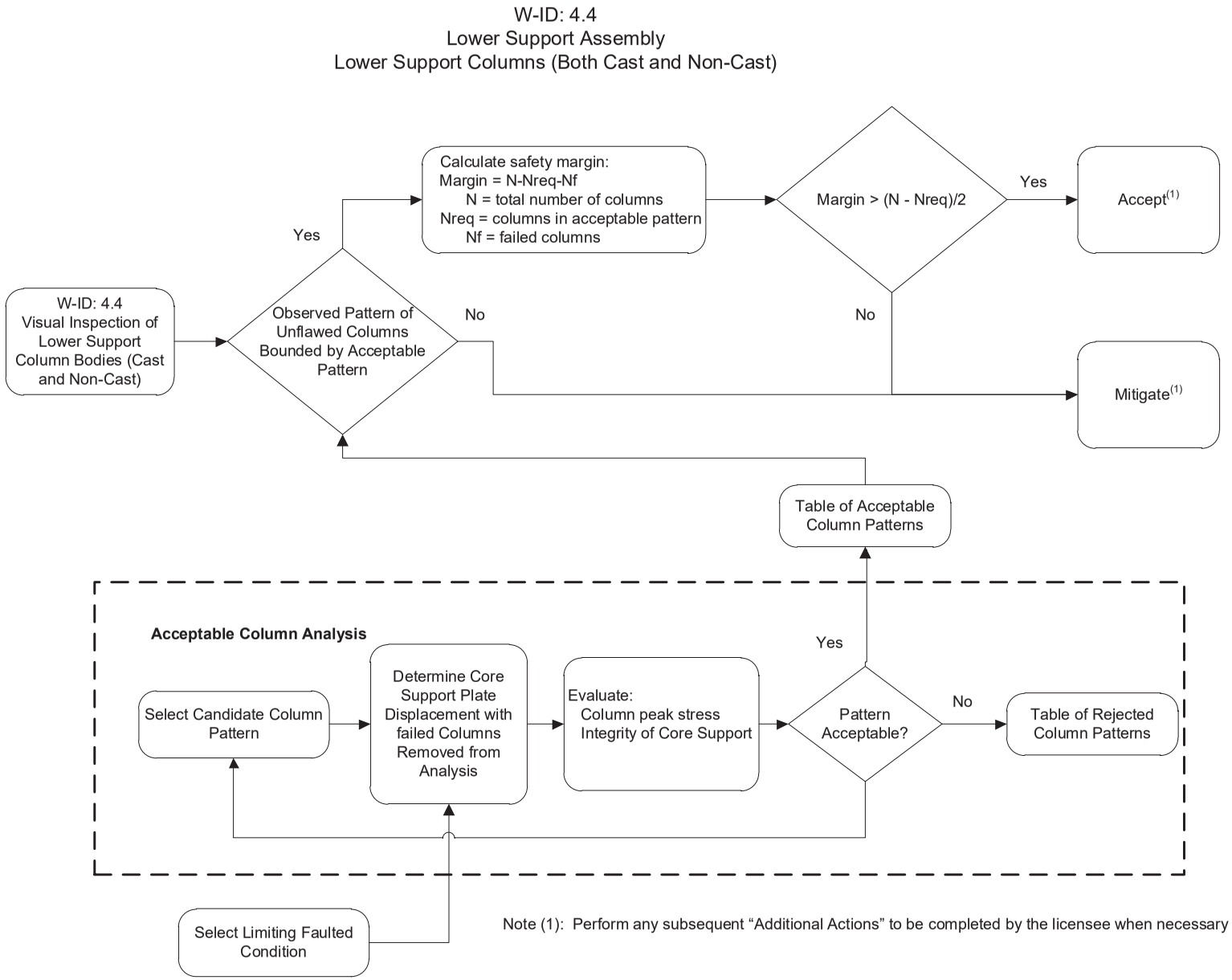
W-ID: 4.3
Core Barrel Assembly – Lower Axial Welds




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



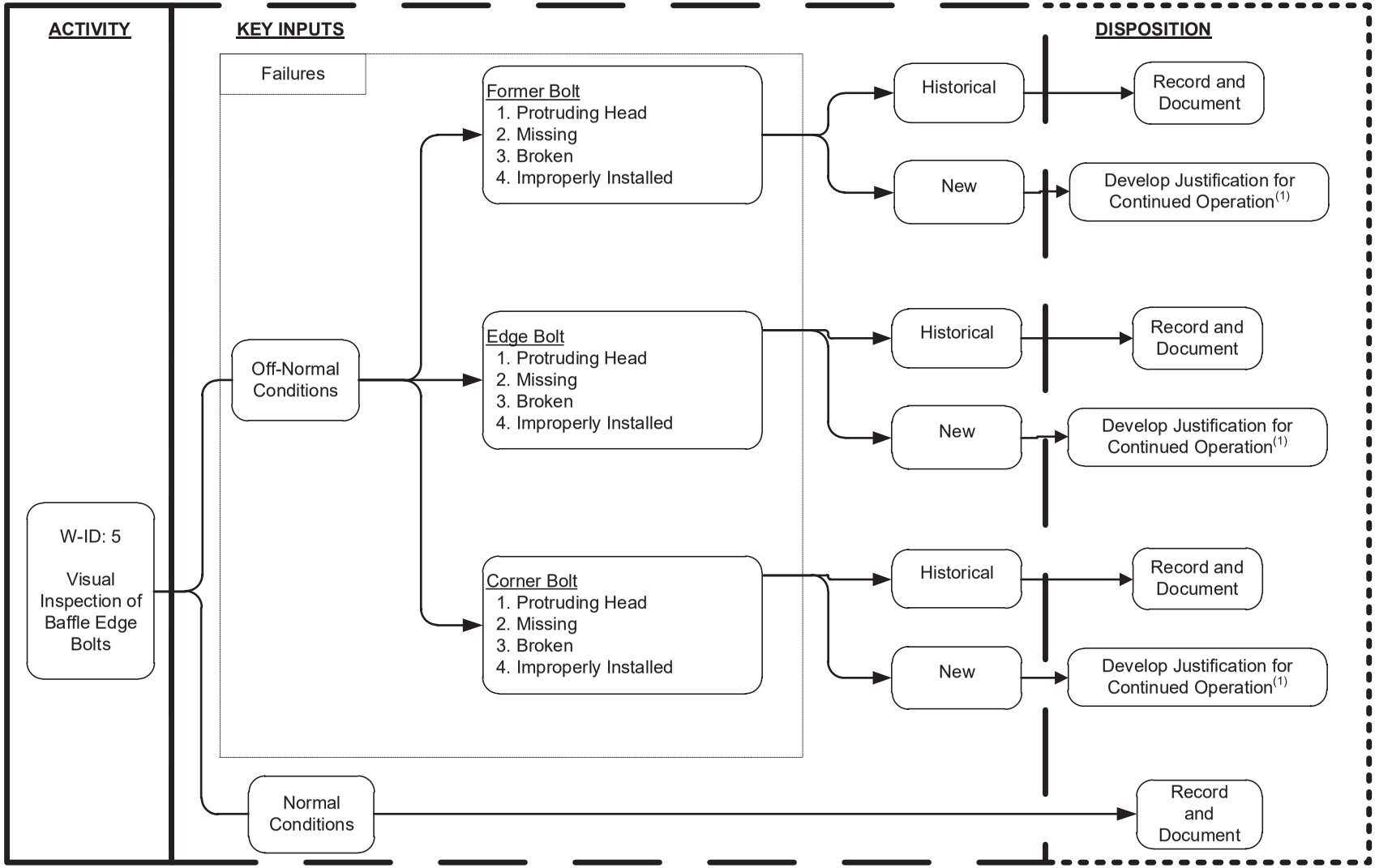
*** This record was final approved on 08/31/2023 13:45:20. (This statement was added by the PRIME system upon its validation)






*** This record was final approved on 08/31/2023 13:45:20. (This statement was added by the PRIME system upon its validation)

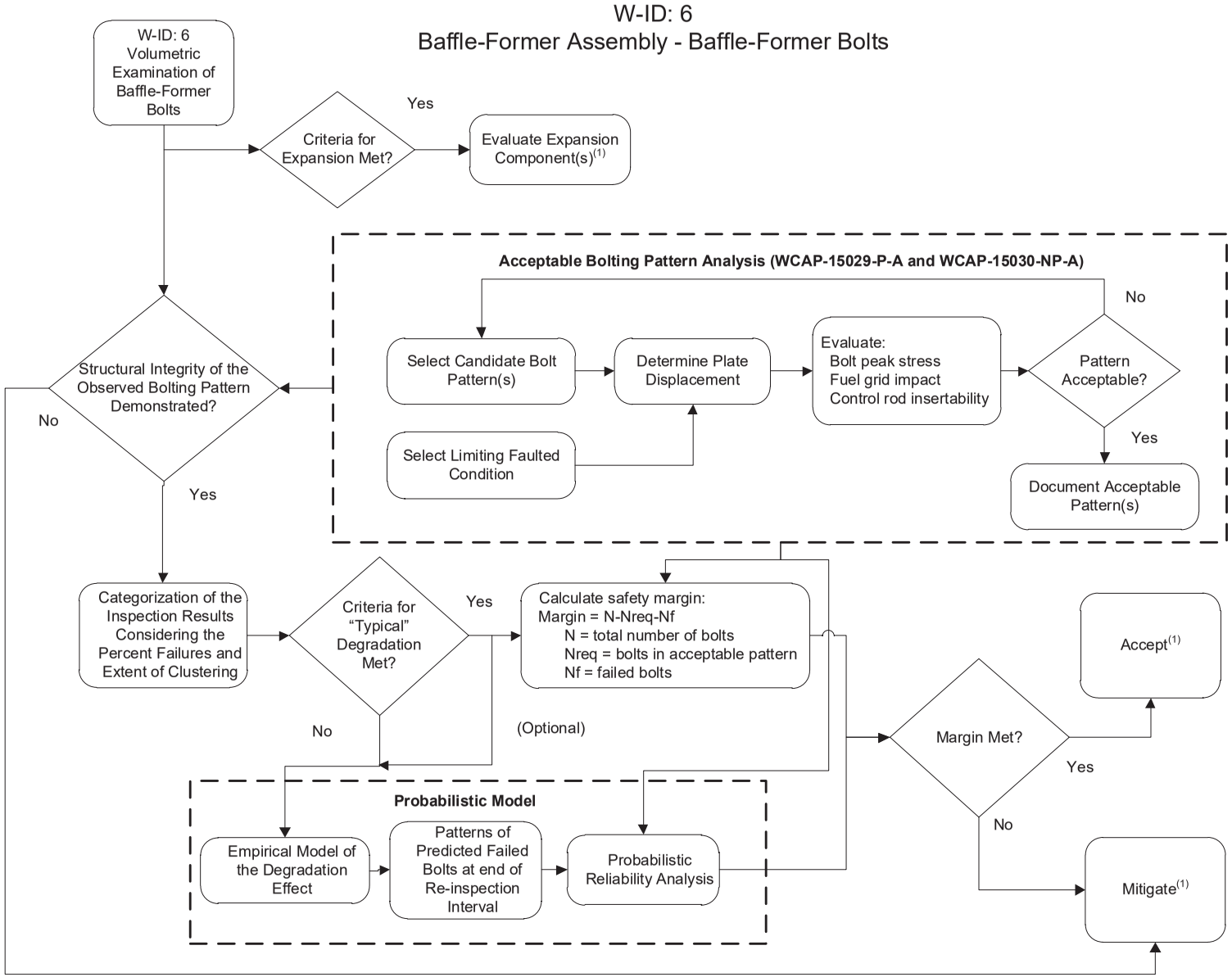
W-ID: 5
Baffle-Former Assembly - Baffle Edge Bolts
FMEA Layout for Bolts (Example)




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



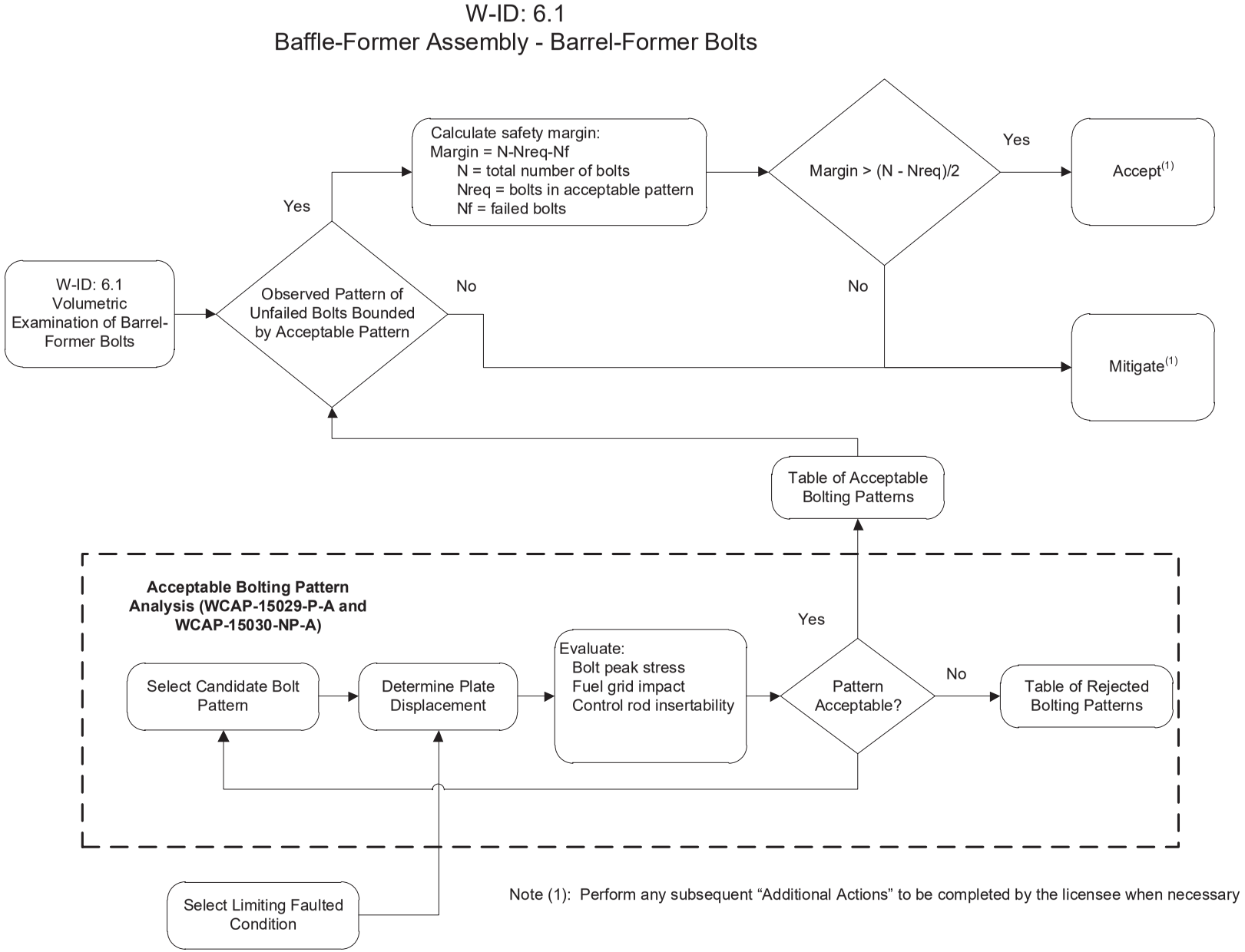
*** This record was final approved on 08/31/2023 13:45:20. (This statement was added by the PRIME system upon its validation)




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



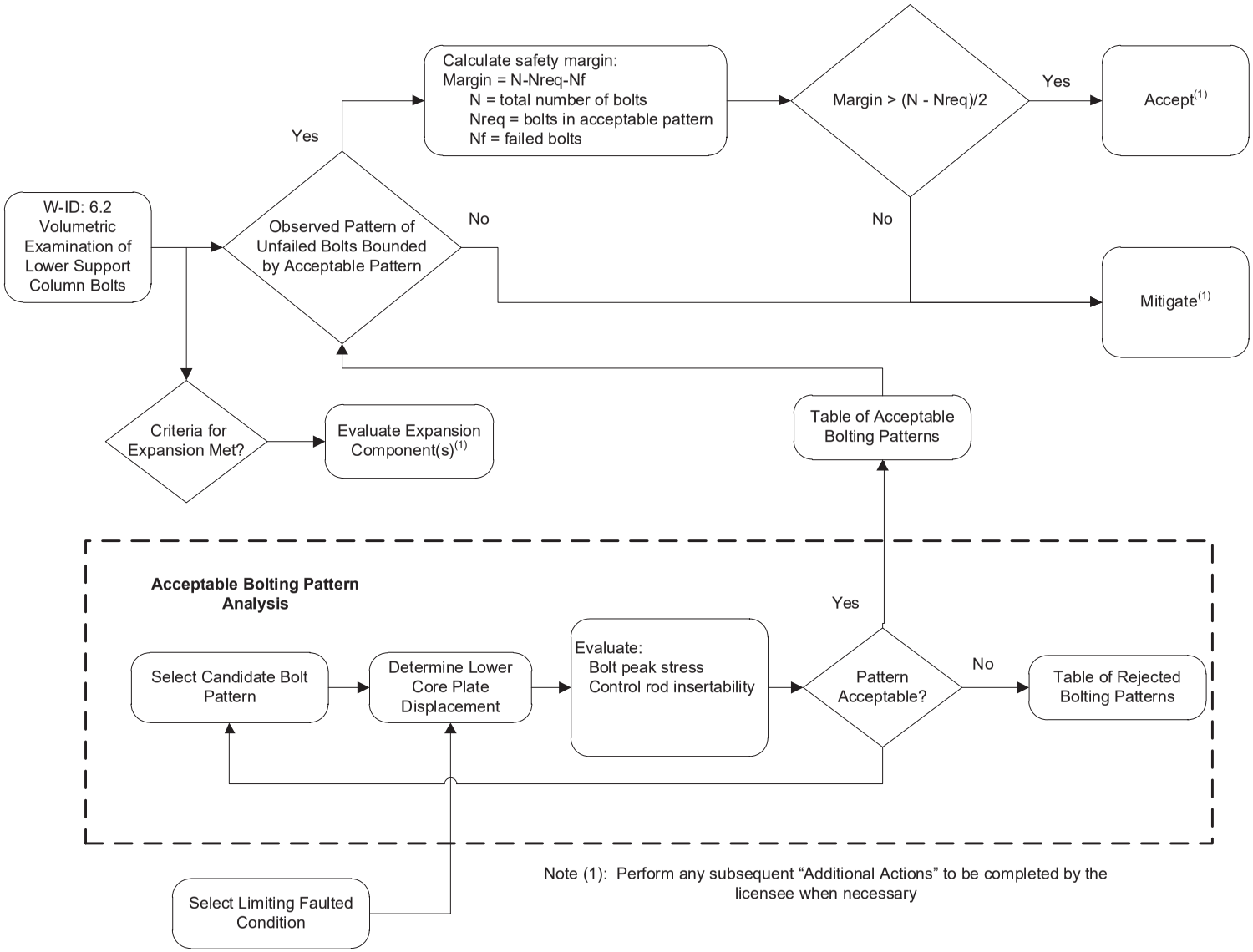
*** This record was final approved on 08/31/2023 13:45:20. (This statement was added by the PRIME system upon its validation)






*** This record was final approved on 08/31/2023 13:45:20. (This statement was added by the PRIME system upon its validation)

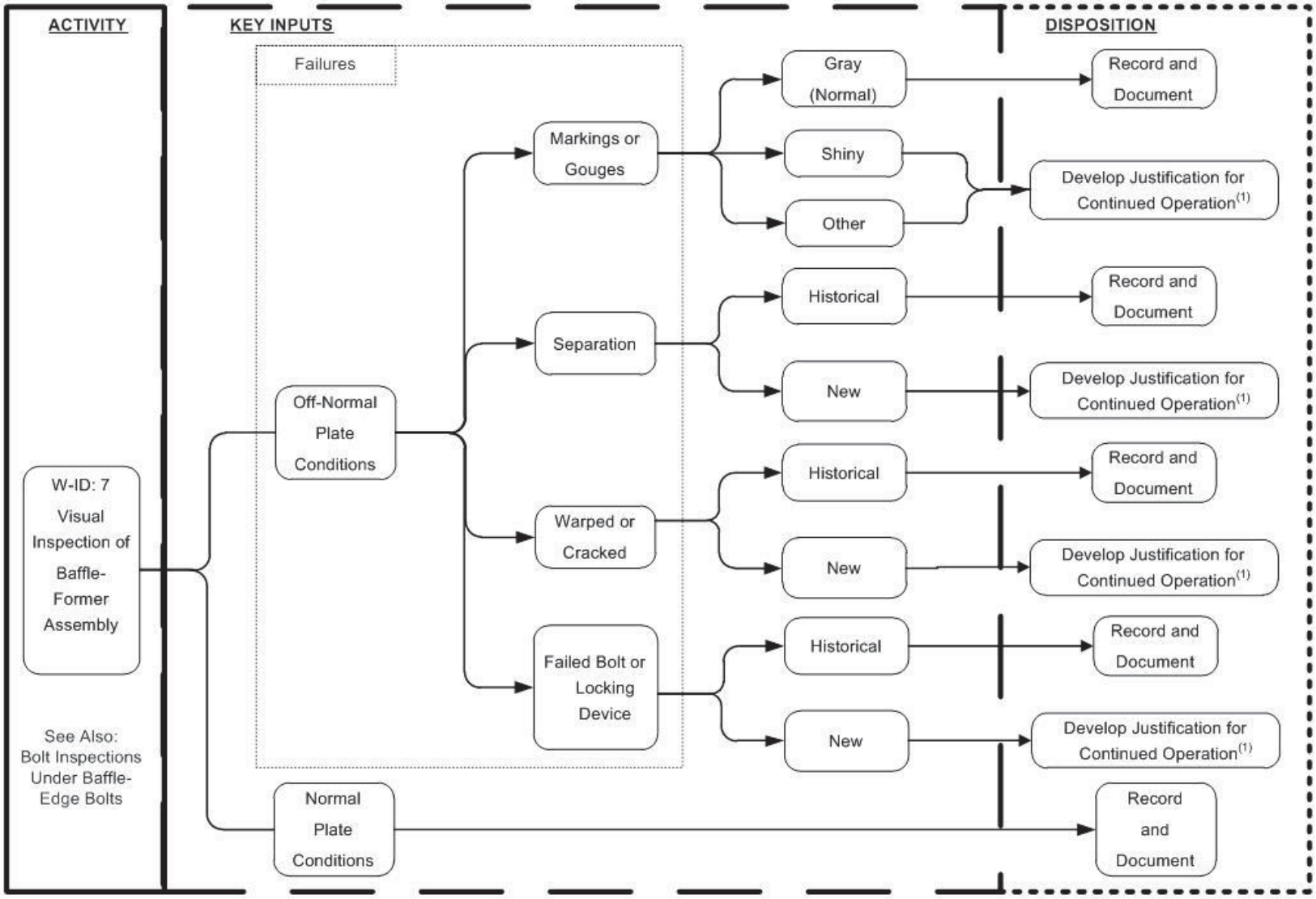
W-ID: 6.2
Lower Support Assembly - Lower Support Column Bolts






*** This record was final approved on 08/31/2023 13:45:20. (This statement was added by the PRIME system upon its validation)

W-ID: 7
Baffle-Former Assembly - FMEA Layout (Example)

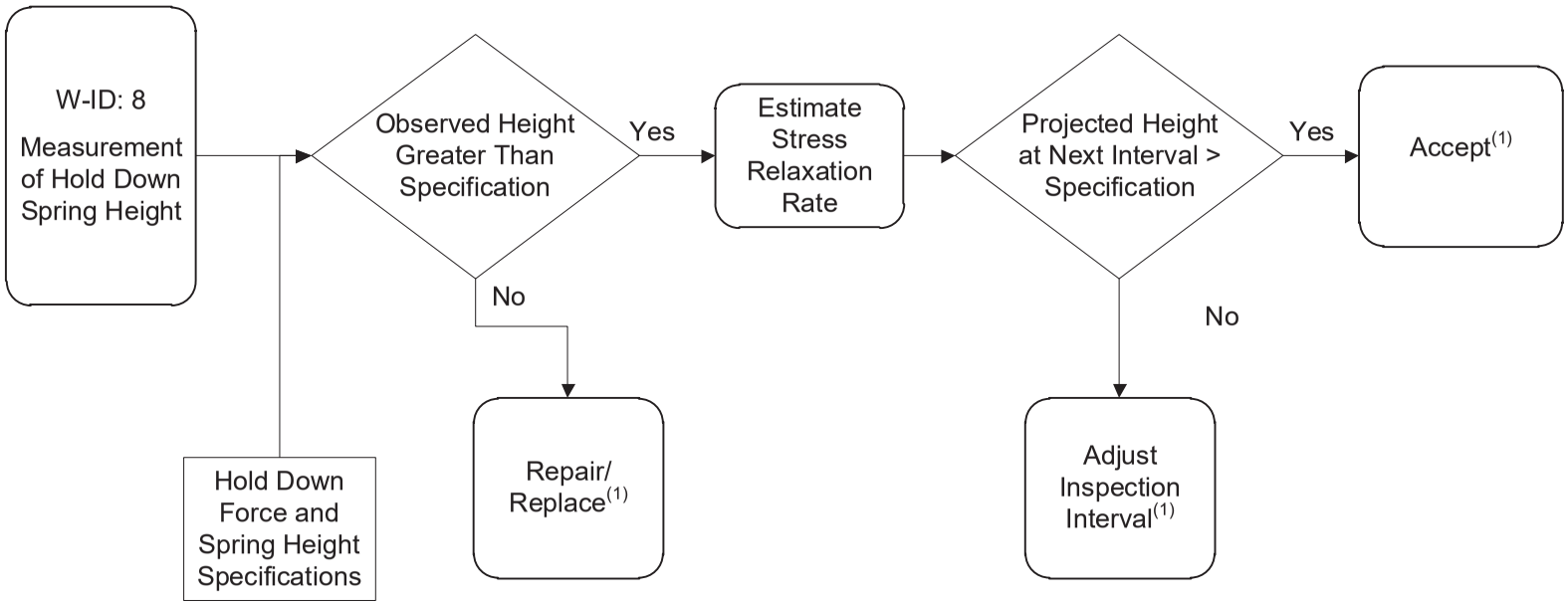



Note (1): Perform any subsequent "Additional Actions" to be completed by the licensee when necessary



*** This record was final approved on 08/31/2023 13:45:20. (This statement was added by the PRIME system upon its validation)

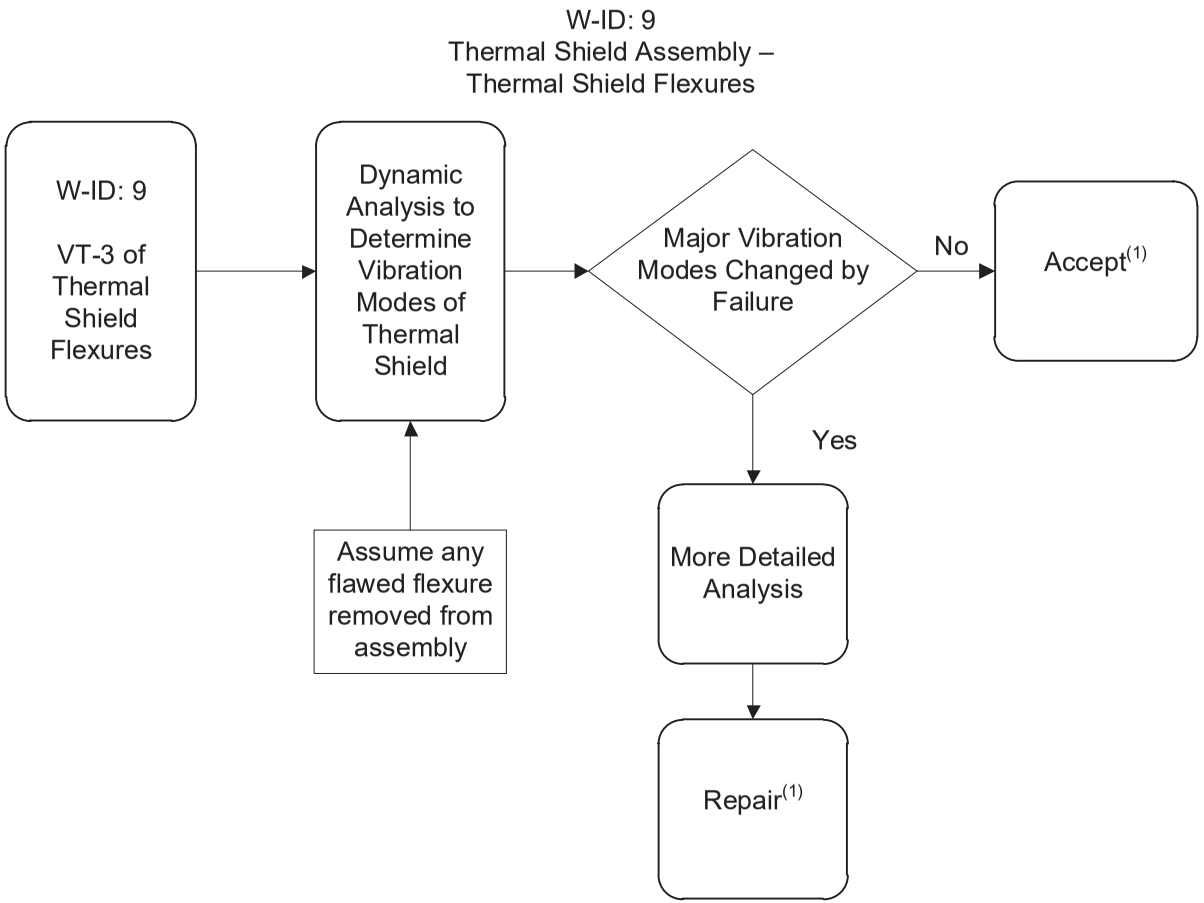
W-ID: 8
Alignment and Interfacing Devices
Internals Hold Down Spring




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



*** This record was final approved on 08/31/2023 13:45:20. (This statement was added by the PRIME system upon its validation)




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



*** This record was final approved on 08/31/2023 13:45:20. (This statement was added by the PRIME system upon its validation)

# APPENDIX G
# PWROG CORRESPONDENCE AND RESPONSES TO NRC REQUESTS FOR ADDITIONAL INFORMATION

Includes the following correspondence:

PWR Owners Group Letter OG-19-164, "Transmittal of WCAP-17096-NP, Revision 3," Reactor Internals Acceptance Criteria Methodology and Data Requirements," PA-MSC-1567, July 31, 2019…….. page G-2

PWR Owners Group Letter OG-22-19, "Transmittal of the Response to Request for Additional Information, RAIs Associated with WCAP-17096, Revision 3, 'Reactor Internals Acceptance Criteria Methodology and Data Requirements,' (PA-MSC-1567)," January 14, 2022…….. page G-5

Westinghouse Letter, LTR-AMLR-20-101, Rev. 1, "Responses to Requests for Additional Information 1, 4, 5, 6, 7, 8, 10, 11 and 14 for WCAP-17096-NP, Rev. 3," January 13, 2022……. page G-8

Westinghouse Letter, LTR-AMLR-21-8, Rev. 0, "Responses to NRC Request for Additional Information 13 for WCAP-17096-NP, Rev. 3," January 13, 2022……. page G-61

Framatome Technical Report, ANP-3104-003-Q1-000, "Framatome Responses to NRC WCAP-17096, Revision 3 RAIs," November 2021……. page G-81



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40TH ANNIVERSARY
PWROG
PWR Owners Group

**Program Management Office**
20 International Drive
Windsor, Connecticut 06095

WCAP-17096-NP, Revision 3
Project Number 99902037

July 31, 2019

OG-19-164

U.S. Nuclear Regulatory Commission
Document Control Desk
11555 Rockville Pike
Rockville, MD 20852

Subject: PWR Owners Group
**Transmittal of WCAP-17096-NP, Revision 3, "Reactor Internals Acceptance Criteria Methodology and Data Requirements," PA-MSC-1567**

The purpose of this letter is to transmit Pressurized Water Reactor Owners Group (PWROG) Topical Report (TR), WCAP-17096-NP, Revision 3, "Reactor Internals Acceptance Criteria Methodology and Data Requirements," in accordance with the Nuclear Regulatory Commission (NRC) TR program for review and acceptance for referencing in regulatory actions (Enclosure 1).

## Topical Report Summary

The subject topical report (TR) defines the methodology and data requirements necessary to evaluate MRP-227 inspection results. MRP-227, "Materials Reliability Program: Pressurized Water Reactor Internals Inspection and Evaluation Guidelines," were developed by the Materials Reliability Program (MRP) to manage the long-term aging of reactor vessel internal components. The NRC issued the Final Safety Evaluation (FSE) for Revision 2 of WCAP-17096-NP-A in May 2016 (ADAMS Accession No. ML16061A243). WCAP-17096-NP-A, Revision 2 contains the methodologies and data requirements that are used to evaluate the MRP-227-A, Rev. 0 inspection results. The NRC issued the FSE for MRP-227-A, Rev. 0 in December 2011. Revision 3 of the WCAP-17096-NP contains the methodologies and data requirements that will be used to evaluate the MRP-227, Rev. 1 inspection results. The NRC issued the FSE for MRP-227-A, Rev. 1 in May 2019 (ADAMS Accession No. 19081A001).

A summary of the changes contained in WCAP-17096, Revision 3 includes:

- Clarification that component methodologies apply to both the initial and subsequent license renewal
- Inclusion of some additional component methodologies for B&W plants
- Realignment of the component categorization for CE and Westinghouse plants due to the primary and expansion component category changes in MRP-227, Revision 1
- Incorporation of the Conditions in the NRC FSE for WCAP-17096-NP-A, Rev. 2
- Replaced the content in WCAP-17096-NP-A, Rev. 2 that was contained MRP-227, Rev. 0 with references to MRP-227, Rev. 1



****This record was final approved on 8/3/2023 23:36:20. (This statement was added by the PRIME system upon its validation)

- Incorporated NEI 03-08 "needed" guidance for Westinghouse baffle-former bolt and guide card wear methodologies

- Incorporated lessons learned from the core barrel weld operating experience into the B&W, CE, and Westinghouse core barrel weld methodologies

It is requested that the NRC review of WCAP-17096-NP-A, Rev. 3 be limited to the specific changes that were made to Revision 2. The specific changes that were made to Revision 2 are identified in the change summary contained in Section 2.4 of the TR and further identified throughout the TR using change bars in the left margin.

**Limits of Applicability**

WCAP-17096-NP, Revision 3 is applicable to all Westinghouse Nuclear Steam Supply System (NSSS) plants, Combustion Engineering (CE) NSSS plants, and Babcock and Wilcox (B&W) NSSS plants.

**Intended Application**

Licensees will use the methodologies contained in WCAP-17096-NP, Revision 3 to evaluate potential degradation of reactor internals. This TR provides a consistent basis for the industry and NRC to evaluate conditions that may occur during MRP-227 inspections.

**Industry Implementation**

WCAP-17096-NP, Revision 3 can be implemented by all U.S. PWRs as discussed in the Limits of Applicability section above.

**Specialized Resource Availability**

The methodologies contained in this TR were developed by the original equipment manufacturer (OEM) vendors for use by the licensees to evaluate the potential material degradation mechanisms in reactor internals. NRC review and approval of the generic TR provides a consistent basis for the evaluation of reactor internal degradation, which are intended to reduce the burden on both licensee and NRC resources for the disposition of plant specific MRP-227 inspection results.

**NRC Review Schedule**

The PWROG requests that the NRC complete their review of the TR within one year.

**Review Fee Waiver**

WCAP-17096-NP, Revision 3 contains a methodology that will be used to disposition MRP-227 Rev. 1 inspection results. The NRC issued a fee waiver for MRP-227, Rev. 1. Therefore, the PWROG will be requesting a fee waiver in accordance with 10 CFR 170.11 (a)(1)(ii). The PWROG will pay for the review fees associated with the review of WCAP-17096-NP, Revision 3 during the NRC review of the fee waiver request.



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This letter transmits one copy of WCAP-17096-NP, Revision 3 (Enclosure 1).

Correspondence related to this transmittal should be addressed to:

Mr. W. Anthony Nowinowski, Program Manager
PWR Owners Group, Program Management Office
Westinghouse Electric Company
1000 Westinghouse Drive
Suite 386
Cranberry Township, Pennsylvania, 16066

If you have any questions, please do not hesitate to contact me at (805) 545-4328 or Mr. W. Anthony Nowinowski, Program Manager of the PWR Owners Group, Program Management Office at (412) 374-6855.

Sincerely yours,

Ken Schrader, Chief Operating Officer and Chairman
PWR Owners Group

KS:WAN:ram

Enclosure 1: One copy of PWROG-17096-NP, Revision 3

cc: PWROG Management Committee
PWROG Materials Committee
PWROG Steering Committee
PWROG Licensing Committee
PWROG PMO
J. Drake, US NRC
J. Andrachek, Westinghouse
D. Radonovich, Westinghouse
B. Wilson, Westinghouse
N. Szweda, Westinghouse
M. Fici, Westinghouse
J. McKinley, Westinghouse
R. Stewart, Framatome
S. Fyfitch, Framatome
R. Hosler, Framatome
S. Davidsaver, Framatome
G. Troyer, Framatome


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**Program Management Office**
1000 Westinghouse Drive, Suite 380
Cranberry Township, PA 16066

PWROG-17096-NP, Revision 3
Docket 99902037
Project 694

January 14, 2022

OG-22-19

U.S. Nuclear Regulatory Commission
Document Control Desk
11555 Rockville Pike
Rockville, MD 20852

Subject: PWR Owners Group
**Transmittal of the Response to Request for Additional Information, RAIs Associated with WCAP-17096, Revision 3, "Reactor Internals Acceptance Criteria Methodology and Data Requirements" (PA-MSC-1567)**

References:

1. Letter OG-19-164, Transmittal of WCAP-17096-NP, Revision 3, "Reactor Internals Acceptance Criteria Methodology and Data Requirements," PA-MSC-1567, dated July 31, 2019

2. Email from the NRC (Fields) to the PWROG (Holderbaum), PWROG-17096 Request For Additional Information, dated September 21, 2020

3. Letter OG-21-81, Transmittal of the Response to Request for Additional Information, RAIs 1-12 and 14 Associated with WCAP-17096, Revision 3, "Reactor Internals Acceptance Criteria Methodology and Data Requirements" (PA-MSC-1567), dated April 8, 2021

4. Email from the PWROG (Holderbaum) to NRC (Fields), PWROG-17096 Draft RAIs Associated with WCAP-17096 - Revision 3, "Reactor Internals Acceptance Criteria Methodology and Data Requirements", dated May 18, 2021

On July 31, 2019, in accordance with the Nuclear Regulatory Commission (NRC) Topical Report (TR) program for review and acceptance, the Pressurized Water Reactor Owners Group (PWROG) requested formal NRC review and approval of WCAP-17096, Revision 3 for referencing in regulatory actions (Reference 1).

The NRC Staff issued a Request for Additional Information (RAI) in an email dated September 21, 2020 (Reference 2).

Responses to RAIs 1 through 12, and 14 and red line markups of the changes to the WCAP associated with the RAI responses were transmitted to the NRC Staff on April 8, 2021, by Reference 3.



*** This record was final approved on 08/31/2023 13:45:20. (This statement was added by the PRIME system upon its validation)

Draft responses to RAI 13 and red line markups of the changes to the WCAP associated with the RAI responses were sent to the NRC Staff by email on May 18, 2021 (Reference 4).

A follow-up call was held with the NRC on October 13, 2021, to discuss the RAI responses that needed to be revised. The responses to RAIs 7, 8, 10, 11 and 13, and the red line markups of the changes to the WCAP associated with the RAI responses that were previously transmitted/sent by References 3 and 4 were revised as discussed on the call. The revisions to these RAI responses are shown and Track Changes, and the revisions to the red line markups of the WCAP are identified with revision bars.

Enclosures 1-3 to this letter contain the final responses to NRC RAIs 1-14 (Reference 2) and the red line markups of the changes to the WCAP associated with the RAI responses.

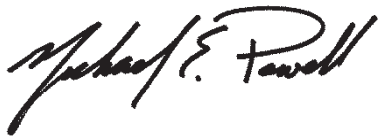
RAI 2, 3, 9, and 12: Framatome only
RAI 1, 4 and 13: Shared, Westinghouse lead author
RAI 5-8, 10-11 and 14: Westinghouse only

Correspondence related to this transmittal should be addressed to:

Mr. W. Anthony Nowinowski, Executive Director
PWR Owners Group, Program Management Office
Westinghouse Electric Company
1000 Westinghouse Drive
Cranberry Township, PA 16066

If you have any questions, please do not hesitate to contact me at (434) 832-2382 or Mr. W. Anthony Nowinowski, Program Manager of the PWR Owners Group, Program Management Office at (412) 374-6855.

Sincerely yours,



Michael Powell
Chief Operating Officer & Chairman
Pressurized Water Reactor Owners Group

cc: PWROG PMO
PWROG Management Committee
PWROG MSC
L. Fields, US NRC
J. Andrachek, Westinghouse
D. Radonovich, Westinghouse
B. Wilson, Westinghouse



*** This record was final approved on 08/31/2023 13:45:20. (This statement was added by the PRIME system upon its validation)

M. Palamara, Westinghouse
K. Musser, Westinghouse
R. Stewart, Framatome
R. Hosler, Framatome
S. Davidsaver, Framatome

Enclosure 1: LTR-AMLR-20-101, Revision 1 (nonproprietary), Responses to NRC Requests for Additional Information 1, 4 – 8, 10, 11 and 14 for WCAP-17096-NP, Rev. 3
Enclosure 2: LTR-AMLR-21-8, Revision 0 (nonproprietary) Responses to NRC Request for Additional Information 13 for WCAP-17096-NP, Rev. 3
Enclosure 3: ANP-3104Q1, Revision 3 (nonproprietary), Framatome Responses to NRC WCAP-17096, Revision 3 RAIs



*** This record was final approved on 08/31/2023 13:45:20. (This statement was added by the PRIME system upon its validation)

Westinghouse

To: James P. Molkenthin
cc:

Date: January 13, 2022

From: Materials and Aging Management
Tel: (412) 374-5559
Fax: (724) 940-8559

Our ref: LTR-AMLR-20-101, Rev. 1

Subject: **Responses to NRC Requests for Additional Information 1, 4, 5, 6, 7, 8, 10, 11 and 14 for WCAP-17096-NP, Rev. 3**

References:
1. U.S. Nuclear Regulatory Commission Letter, "U.S. NRC Request for Additional Information for Pressurized Water Reactor Owner's Group Topical Report WCAP-17096-NP, Revision 3, 'Reactor Internals Acceptance Criteria Methodology and Data Requirements'," September 21, 2020.
2. Westinghouse Report, WCAP-17096-NP, Rev. 3, "Reactor Internals Acceptance Criteria Methodology and Data Requirements," July 29, 2019 (ADAMS Accession No. ML19218A179).
3. Westinghouse Letter, LTR-AMLR-20-101, Rev.0, "Responses to NRC Requests for Additional Information 1, 4 - 8, 10, 11 and 14 for WCAP-17096, Rev. 3," March 30, 2021.

Attachment 1 of Reference 3 provided the responses to the NRC Requests for Additional Information (RAI), except for RAI Number 13, which were issued via Reference 1 on WCAP-17096, Rev. 3 (Reference 2). Attachment 2 of Reference 3 contained the redline markups of the changes to WCAP-17096, Rev. 3 that were discussed in the RAI responses in Attachment 1 of Reference 3.

Revision 1 of this letter revises the responses to RAIs 7, 8, 10 and 11 and the redline markups associated with the revised RAI responses, that were previously transmitted via Reference 3, as discussed in the October 13, 2021 meeting with the NRC Staff. The changes to RAI responses are indicated with revision bars.

Framatome previously provided the responses to RAI Numbers 2, 3, 9, and 12, and will transmit the revised RAI response to RAI 12.

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If you have additional questions or require further information, please contact the undersigned.

Authored by: ELECTRONICALLY APPROVED*
David C. Radonovich
Materials & Aging Management

Verified by: ELECTRONICALLY APPROVED*
Joshua K. McKinley
Materials & Aging Management

Reviewed by: ELECTRONICALLY APPROVED*
James D. Andrachek
PWR Owners Group PMO

Approved by: ELECTRONICALLY APPROVED*
Kaitlyn M. Musser, Manager
Materials & Aging Management

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# ATTACHMENT 1

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



***This record was final approved on 10/4/2023 3:23:48 PM. (This statement was added by the PRIME system upon its validation)

**BACKGROUND:**

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

**NOTES:**

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

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

**Request for Additional Information (RAI) 01**

Applicable TR Sections, Appendices, Tables, Figures, or Pages:
TR Section 1 (TR Page 1-1)

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

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

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

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

**Response to RAI 01:**

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



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

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

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

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

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

Reference 36 of the TR will be revised to reflect the NRC-approval of MRP-227, Rev. 1-A as shown on page 6 of Attachment 2.



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**RAI 04**

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

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

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

## Response to RAI 04:

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

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

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

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



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

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

The ASME BPVC published Code Case N-889 [24] in July 12, 2018, which contains the latest reference stress corrosion cracking growth rate curves for Irradiated Austenitic Stainless Steel in Light-Water Reactor Environments (both PWR and BWR environments are considered). The technical bases of this Code Case are per ASME Code Committee Record 17-506 [15], along with Reference 46 of the TR. The NRC issued the conditional acceptance of this Code Case in Draft RG 1.147 Revision 20 (ADAMS Accession No. ML20120A631) [25], which was issued for public comment. The NRC specified several conditions for the use of Code Case N-889. The final NRC approval of this Code Case is scheduled for 2022. Based on the conditional approval of N-889 and the NRC comments in Draft RG 1.147 Rev. 20, that the following revisions to WCAP-17096, Revision 3 will be made:

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

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

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

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

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

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



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*Disposition Curves Application,"* (TR Reference 46), will be used. The acceptance criteria developed for fluence levels greater than $1.4x10^{22}$ n/cm$^2$ (E>1MeV) (20 dpa) must be approved by the NRC as discussed in Draft RG 1.147, Rev. 20 [25].

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

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

| | | |
|---|---|---|
| Section 2.2 | Page 2-6 | See page 2 of Attachment 2 |
| Appendix C | Page C-9 | See page 15 of Attachment 2 |

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

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



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**NRC RAI 05**

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

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

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

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

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

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

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

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



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## Response to RAI 05:

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

The text in the submittal requirement in Section 2.2 of the TR will be revised to state that it applies to the functionality analysis of the Ribs and Rings component. These changes are shown on page 3 of Attachment 2. The submittal requirement was added to CE-ID: 3.2 as shown on page 17 of Attachment 2.



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**NRC RAI 06**

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

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

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

NRC Request:
(a) Justify why TR Section 2.3 does not incorporate the above 2019 Interim Guidance for WEC core barrel assembly and CE core support assembly MAWs and LAWs, as provided in EPRI Letter Nos. MRP 2019-023 and MRP 2019-009.

(b) Justify why the acceptance criteria and data analysis methods TR Appendix C, CE-ID Items 6.1 and 6.2 and *TR* Appendix E, W-ID Items 4.2 and 4.3 do not include acceptance criteria and data analysis methods for the one-time VT-3 inspection of the MAWs and LAWs for WEC core barrels and CE core support barrels.

## Response to RAI 06:

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

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

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

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

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



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

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



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**NRC RAI 07**

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

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

Background:

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

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

Issue:

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

NRC Request:

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

Issue:

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



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NRC Request:

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

Issue:

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

NRC Request:

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

## Response to RAI 07:

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

These changes are shown by the removal of CE-ID: 1, CE-ID: 1.1, and CE-ID: 1.2 from the list of CE components on page 8 of Attachment 2, in the removal of CE-ID: 1 for Core Shroud Bolts on pages 9 and 10 of Attachment 2, in the removal of CE-ID: 1.1 for Core Support Column Bolts on pages 11 and 12 of Attachment 2, and also in the removal of CE-ID: 1.2 for Barrel-Shroud Bolts on pages 13 and 14 of Attachment 2. The flowcharts for CE-ID: 1, CE-ID: 1.1, and CE-ID: 1.2 were removed as shown on pages 21-23.

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



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A statement will be added to W-ID: 6 for Baffle-Former Bolts, that PWROG-18034-P-A [20] and PWROG-18034-NP-A [21] supplement the methodology (WCAP-15029-P-A [18] and WCAP-15030-NP-A [19]) as shown on page 25 of Attachment 2.

This same change will also be made for W-ID: 6.1 for Barrel-Former Bolts, as shown on pages 27 and 28 of Attachment 2. PWROG-18034-P-A and PWROG-18034-NP-A will also be added as references as shown on page 7 of Attachment 2.

(c) Section 1.1 of the TR states:

"There are no specific NEI 03-08 Mandatory, Needed, or Good Practice elements in this document. However, the methodologies defined herein can be use used to meet the following NEI 03-08 Needed requirement in Section 7.5 of MRP-227 [1]:

> 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, WCAP-17096-NP or equivalent methods)".

Therefore, this TR was not intended to establish NEI 03-08 Needed requirements. After NRC approval of the TR, it can be utilized as one of the available options to fulfill the NEI 03-08 Needed requirements in Section 7.5 of MRP-227.

The discussion in Section 3.3 of the TR regarding the confirmation of the plant-specific applicability of a generic ABPA is a process step that is discussed within the W-ID: 6 methodology, rather than Section 3.3 of the TR. Therefore, the statement in W-ID: 6 regarding the approach will be revised as follows to provide this clarification.

The "Approach" in W-ID: 6 currently states:

> "Generic work completed in previous PWROG program."

This statement will be revised to:

> "Generic ABPAs that were previously performed should be confirmed to be applicable on a plant-specific basis."

This change is shown on page 26 of Attachment 2.

In addition, the following statement will be deleted from Section 3.3 of the TR:

> "The plant-specific applicability of the generic acceptable patterns must be confirmed by the user, since certain inputs may have been changed over the two decades since these patterns were originally developed."

This change is shown on page 4 of Attachment 2.



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**NRC RAI 08**

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

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

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

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

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

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

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

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

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

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

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



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## Response to RAI 08:

Background:

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

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

These changes are shown on page 8 of Attachment 2 for the list of CE component methodologies in Appendix C, and the same changes as shown above will be made to the component numbering throughout the TR.

(a) The purpose of Section 3.6 of the TR is to discuss that a fatigue screening (and also SCC evaluation for CE-ID: 7) can be performed in lieu of EVT-1 inspections for three CE components: the CSB Flexure Weld (CE-ID: 7), the Core Support Plate (CE-ID: 9), and the Fuel Alignment Plate (CE-ID: 10). During the NRC review of MRP-227, Rev. 1-A, the Staff issued RAI 16, which included two parts, Part (a) was associated with the justification of the criteria that are used for screening for fatigue, and Part (b) was associated with how fatigue screening accounts for the contribution of SCC for CE-ID: 7.

In response to the RAI 16 for MRP-227, Rev. 1-A, EPRI discussed the option to perform fatigue screening for the three CE components (CE-ID: 7, 9, and 10) in lieu of EVT-1 inspections as stated in the "Examination Method / Frequency" column of Table 4-2 in MRP-227, Rev. 1-A [6] as follows:

> *"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."*

The CSB Flexure Weld (CE-ID: 7) includes an additional requirement to consider SCC in Table 4-2 of MRP-227, Rev. 1-A as follows:

> "*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."*

Therefore, the requirements for the fatigue screening for all three CE components, and the additional requirement to perform SCC evaluation for the CSB Flexure Weld, are defined in MRP-227, Rev. 1-A. The methodologies defined in this TR are intended to be used to develop acceptance criteria to disposition indications from the EVT-1 examinations, if the EVT-1 inspections are performed.

In the NRC Safety Evaluation for MRP-227, Rev. 1-A, the Staff stated the following regarding Item (a) of RAI 16:

> *"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."*



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The Staff made the following statement regarding Item (b) of RAI 16:

> *"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."*

Therefore, the fatigue screening and SCC evaluation are outside of the scope of this TR, and Section 3.6 of this TR will be deleted. This change is shown on page 5 of Attachment 2.

(b) As discussed in Part (a) of this RAI response, the requirements for the fatigue screening for all three CE components, as well as the requirement to perform the SCC evaluation of the CSB Flexure Weld, are defined in MRP-227, Rev. 1-A. Therefore, there is no need to discuss the requirements within the inspection methodologies of CE-ID: 7, CE-ID: 9, and CE-ID: 10. This is consistent with one of the objectives of Rev. 3 of the TR that is stated in the last paragraph of Section 1:

> *"An additional objective of Revision 3 of this document is to replace the content repeated in WCAP-17096 from MRP-227 with references to MRP-227. This change provides simplification, clarifies which requirements are defined within MRP-227, and avoids inadvertent application of out-of-date information."*

Therefore, the discussion of the requirements that were identified in Part (a) of this RAI response will be deleted from the "Inspection Method" of CE-ID: 7, CE-ID: 9, and CE-ID: 10. The "Inspection Method" for these components will be revised to: "See MRP-227 Table 4-2." These changes are shown on pages 18, 19, and 20 of Attachment 2.

(c) As previously discussed in Part (a) of the RAI response, the requirement for the fatigue screening was discussed in the "Examination Method / Frequency" column of Table 4-2 in MRP-227, Rev. 1-A [6]. The Staff acknowledged that the MRP-227, Revision 1-A fatigue screening evaluations for CE-ID: 7, CE-ID: 9, and CE-ID: 10 are plant-specific in Section 3.1.3.4 of the Safety Evaluation for MRP-227, Rev. 1-A:

> *"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."*

Therefore, the response to Part (c) of this RAI was addressed in MRP-227, Rev. 1-A, and the methodology for these plant-specific fatigue screening evaluations is outside of the scope of this TR.



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**NRC RAI 10**

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

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

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

## Response to RAI 10:

MRP-211, Revision 1 [22] (Materials Reliability Program: PWR Internals Age-Related Material Properties, Degradation Mechanisms, Models, and Basis Data – State of Knowledge) was issued in 2017. MRP-211 discusses the current state of knowledge, available relevant data, and technical bases for trend model formulations of key aging mechanisms for long term functionality evaluations, including fracture toughness. MRP-211 concluded that all fracture toughness data are bounded by a saturated value for $K_{JC}$ of 34.6 ksi-√in. MRP-211 further states in the last paragraph of subsection 2.2.4: "this value provides an appropriate lower bound of all fracture toughness data and is recommended by the expert panel for use in engineering evaluations and assessments."

As discussed in the May 27, 2021 meeting with the NRC Staff [26], the fracture toughness data that were used to justify the fracture toughness limits in this TR is being re-evaluated.

The TR will be revised to include MRP-211 as a reference as shown on page 7 of Attachment 2, and note 2 will be added to the tables that include the 34.6 ksi-√inch fracture toughness values, as shown on page 16 of Attachment 2 for CE-ID: 2.

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

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



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



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**NRC RAI 11**

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

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

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

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

## Response to RAI 11:

Section 3 of the TR is intended to review the types of inspections and evaluations that are required in MRP-227, Rev. 1-A [6]. Specific visual inspections are covered in Section 3.2 for "General Condition" (i.e. VT-3) examinations and Section 3.4 for more detailed "Visual Cracking" (i.e. EVT-1 or VT-3) examinations. Section 3.5 "Visual Other" covers VT-1 inspections that are used to inspect for potential void swelling of the welded CE core shroud.

The discussion in the 2nd paragraph in Section 3.5 of the TR provides a status on the functionality analyses that was performed in MRP-230 [23] (Reference 6 in the TR) regarding void swelling, and to identify that additional sensitivity studies could be performed. The TR does not contain a requirement to perform additional sensitivity studies; however, the current wording could be interpreted that the additional sensitivity studies are required. If void swelling is observed, actions to address this condition would be addressed in the licensee's corrective action program. The TR will be revised as discussed below.

The 2nd paragraph in Section 3.5 currently states:

"These examinations are meant to provide an early warning of swelling in the structure. A functionality evaluation was performed in MRP-230 [6], which indicates that gap openings may indicate void swelling. Evaluation of this swelling-related data will require additional sensitivity studies to relate the swelling level to the predicted distortion and gap opening in the structure. Sensitivity studies may be necessary to



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evaluate the current condition, predict future effects, and determine aging management strategies, which are beyond the scope of this document."

The 2nd paragraph in Section 3.5 will be revised as follows:

"These examinations are intended to provide an early indication of swelling in the structure. A functionality evaluation was performed in MRP-230 [6], which shows that gap openings may indicate void swelling. Additional sensitivity studies could be performed to correlate the void swelling level to the predicted distortion and gap opening in the structure, which would be used to evaluate the current condition, predict future effects, and determine aging management strategies. Void swelling, if identified, will be addressed in the licensee's corrective action program."

The revision to the 2nd paragraph in Section 3.5 is shown on page 5 of Attachment 2.



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**NRC RAI 14**

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

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

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

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

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

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

## Response to RAI 14:

Yes, there is a typographical error in the data requirements section of W-ID: 1 for the Guide Plates (Cards). The revised bulleted items for the "Data Requirements" section are shown on page 24 of Attachment 2.

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



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## References

1. NRC NUREG-2191, Volume 2, "Generic Aging Lessons Learned for Subsequent License Renewal Applications," Chapter XI.M16A, *PWR Vessel Internals*, 2017 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML16274A399).
2. Westinghouse Report, WCAP-17096-NP, Revision 3, "Reactor Internals Acceptance Criteria Methodology and Data Requirements," July 2019 (ADAMS Accession No. ML19218A179).
3. Westinghouse Report, WCAP-17096-NP-A, Revision 2 "Reactor Internals Acceptance Criteria Methodology and Data Requirements," August 2016 (ADAMS Accession No. ML16279A320).
4. NRC Correspondence Letter and Safety Evaluation from Mr. Kevin Hseuh (USNRC) to Ms. Anne Demma (EPRI), "Final Safety Evaluation of WCAP-17096-NP, Revision 2, 'Reactor Internals Acceptance Criteria Methodology and Data Requirements,'" May 3, 2016 (ADAMS Accession No. ML16061A243).
5. *Materials Reliability Program: Pressurized Water Reactor Internals Inspection and Evaluation Guidelines (MRP-227-A).* EPRI, Palo Alto, CA: 2011. 1022863. (ADAMS Accession Nos. ML12017A194, ML12017A196, ML12017A197, ML12017A191, ML12017A192, ML12017A195 and ML12017A199).
6. *Materials Reliability Program: Pressurized Water Reactor Internals Inspection and Evaluation Guidelines (MRP-227, Revision 1-A).* EPRI, Palo Alto, CA: 2019. 3002017168. (ADAMS Accession No. ML19339G350).
7. *Models of Irradiation-Assisted Stress Corrosion Cracking of Austenitic Stainless Steels in Light Water Reactor Environments, Volume 2: Disposition Curves Application*. EPRI, Palo Alto, CA, 2014. 3002003103.
8. EPRI Letter No. MRP 2019-023, "Transmittal of MRP-227-A-Related Interim Guidance Regarding PWR Core Barrel," September 3, 2019 (ADAMS Accession ML19249B102; includes enclosure of MRP 2019-009 [RAI Set Ref. 9 below]).
9. EPRI Letter No. MRP-2019-009, "Transmittal of NEI 03-08 "Good Practice" Interim Guidance Regarding MRP-227-A and MRP-227, Revision 1 Core Barrel and Core Support Barrel Inspection Requirements," July 17, 2019.
10. NRC Correspondence Letter and Safety Evaluation from Mr. Dennis Morey (USNRC) to Mr. Brian Burgos (EPRI), "Final Safety Evaluation for Electric Power Research Institute Topical Report MRP-227, Revision 1, 'Materials Reliability Program: Pressurized Water Reactor Internals Inspection and Evaluation Guideline," April 25, 2019 (ADAMS Accession No. ML19081A001).
11. Westinghouse Report, PWROG-18034-NP, Revision 0, "Updates to the Methodology in WCAP-15030-NP-A, Rev. 0, 'Westinghouse Methodology for Evaluating the Acceptability of Baffle-Former-Barrel Bolting Distributions Under Faulted Load Conditions,'" October 2018 (ADAMS Accession No. ML18306A491).
12. Final Safety Evaluation by the Office of Nuclear Reactor Regulation, Topical Report PWROG-18034-P, Revision 0, "Updates to the Methodology in WCAP-15030-NP-A, Rev. 0, 'Westinghouse Methodology for Evaluating the Acceptability of Baffle-Former-Barrel Bolting Distributions Under Faulted Load Conditions,'" May 26, 2020 (ADAMS Accession No. ML20134M356).



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13. Westinghouse Technical Bulletin TB-19-5, "Westinghouse NSSS PWR Thermal Shield Degradation," Westinghouse Non-Proprietary Class 3, October 9, 2019 (ADAMS Accession No. ML19302F228).

14. Westinghouse Document, Presentation to the NRC Staff (Non-Proprietary Version), "Westinghouse Baffle-Former Bolt Predictive Methodology," Josh McKinley and Matt Palamara, September 18, 2019 (ADAMS Accession No. ML19274F472).

15. ASME Code Committee Record #17-506 - Irradiation Assisted Stress Corrosion Crack Growth in Irradiated Materials.

16. U.S. Nuclear Regulatory Commission Letter, "U.S. NRC Request for Additional Information for Pressurized Water Reactor Owner's Group Topical Report, WCAP-17096-NP, Revision 3, 'Reactor Internals Acceptance Criteria Methodology and Data Requirements'," September 21, 2020.

17. Nuclear Energy Institute Document, NEI 03-08, Rev. 4, "Guideline for the Management of Materials Issues," October 2020.

18. Westinghouse Report, WCAP-15029-P-A, Rev. 1, "Westinghouse Methodology for Evaluating the Acceptability of Baffle-Former-Barrel Bolting Distributions Under Faulted Load Conditions," January 1999.

19. Westinghouse Report, WCAP-15030-NP-A, Rev. 0, "Westinghouse Methodology for Evaluating the Acceptability of Baffle-Former-Barrel Bolting Distributions Under Faulted Load Conditions," January 1999.

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

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

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

23. *Materials Reliability Program: Functionality Analysis for Westinghouse and Combustion Engineering Representative PWR Internals (MRP-230 Rev. 2).* EPRI, Palo Alto, CA: 2012. 1021026.

24. ASME Code Case N-889, "Reference Stress Corrosion Crack Growth Rate Curves for Irradiated Austenitic Stainless Steel in Light-Water Reactor Environments," July 12, 2018.

25. Proposed Revision 20 to Regulatory Guide RG 1.147, "Inservice Inspection Code Case Acceptability, ASME Section XI, Division 1," January 2021. *(Attached in PRIME)*

26. PWROG Presentation to the NRC Staff, "PWROG Evaluation of the Potential Non-Conservative Fracture Toughness in BWRVIP-100, Rev. 1-A", May 27, 2021 (ML 21147A012).



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# ATTACHMENT 2

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

## Purpose

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

## References

1. U.S. Nuclear Regulatory Commission Letter, "U.S. NRC Request for Additional Information for Pressurized Water Reactor Owners Group Topical Report, WCAP-17096-NP, Revision 3, 'Reactor Internals Acceptance Criteria Methodology and Data Requirements'," September 21, 2020.
2. Westinghouse Report, WCAP-17096-NP, Rev. 3, "Reactor Internals Acceptance Criteria Methodology and Data Requirements," July 2019 (ADAMS Accession No. ML19218A179).
3. LTR-AMLR-20-101, Rev.0, "Responses to NRC Requests for Additional Information 1, 4 - 8, 10, 11 and 14 for WCAP-17096, Rev. 3," March 30, 2021.



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

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

Condition 2 Resolution

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

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

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

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

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

This statement has been added to the following components:

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



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

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

This statement has been added to the following components:

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

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

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

This statement has been added to the following component:

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

*Group 5 RVI Components*

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

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

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

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

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



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been specified where the first signs of degradation are expected to be visual (e.g., wear). In any case, it is prudent engineering practice to anticipate the range of possible visual observations and define resolution strategies.

A process identifying potential relevant conditions arising from each recommended VT-3 examination and defining resolution strategies is required for each VT-3 examination. This process may take the form of a failure modes and effects analysis (FMEA).

## 3.3 ULTRASONIC TESTING (BOLTS)

Within the I&E Guidelines, ultrasonic testing (UT) examinations are conducted to identify failed bolts. All bolts with positive indications of cracking are assumed to be failed. The bolting systems in the internals are generally highly redundant. Acceptance criteria are based on acceptable bolting patterns that guarantee structural stability through both normal operation and design basis transients. To establish an appropriate inspection interval, the current distribution of unfailed bolts must contain sufficient margin to demonstrate that the number of anticipated failures will not cause the distribution to fall below the minimum pattern. This will require either a historical analysis of bolt failure rates or a detailed model of bolt failure mechanisms.

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

## 3.4 VISUAL CRACKING

Visual examinations to identify cracking are generally recommended where irradiation-assisted stress corrosion cracking (IASCC), intergranular stress corrosion cracking (IGSCC), stress corrosion cracking (SCC), or fatigue is identified as the cracking mechanism. The initial visual examinations in the B&W plants are all based on VT-3 requirements. The VT-3 exams were deemed to be adequate because the structures are relatively flaw tolerant. Appropriate follow-on actions might include EVT-1, UT, or eddy-current testing (ET) exams to determine flaw size. These options would be identified as part of the recommended procedure for resolution of the original VT-3 observation. Consistent with the current state-of-the-art within the Boiling Water Reactor Vessel and Internals Project (BWRVIP), EVT-1 has been identified as the appropriate visual examination procedure for several of the Combustion Engineering (CE) and Westinghouse components. The EVT-1 examination can produce flaw size information that would lead directly to a fracture mechanics evaluation. For each component, the most appropriate visual inspection method has been determined (e.g., EVT-1 or VT-3) based upon the relevant conditions (i.e. flaw tolerance, inspectability, etc.). For components where a visual examination is identified, the specific technique is specified in MRP-227. UT or ET may also be used as enhanced or supplemental examinations, where appropriate.



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

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

### 3.5 VISUAL OTHER

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

These examinations are meant to provide an early warning of swelling in the structure. A functionality evaluation was performed in MRP-230 [6], which shows that gap openings may indicate void swelling. Additional sensitivity studies could be performed to relate the swelling level to the predicted distortion and gap opening in the structure, which would be used to evaluate the current condition, predict future effects, and determine aging management strategies. Void swelling, if identified, will be addressed in the licensee's corrective action program.

### ~~3.6 FATIGUE (QUALIFY BY EVALUATION)~~

~~Three component items in the CE design are included in the list of Primary components due to concerns about fatigue and permit the use of a "fatigue analysis" to address potential fatigue degradation[1]. Due to the plant-specific nature of the evaluation required for license extension programs, fatigue analysis was not included in the MRP engineering evaluation and assessments (i.e., functionality analysis). It is considered a high probability that an evaluation will demonstrate a negligible probability of fatigue crack initiation in these components. However, pending resolution by evaluation[2], these component items are included in the Primary Component list.~~

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

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



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

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

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

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

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

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

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

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

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

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

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

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

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



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

**CE Primary and Expansion Components**

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

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

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

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

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

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

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

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

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

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

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

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

CE-ID: 12 Lower Support Structure – Deep Beams



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C-2

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



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C-3

| | |
|---|---|
| ~~CE ID: 1~~ | ~~Core Shroud Assembly (Bolted)~~<br>~~Core Shroud Bolts~~ |
| ~~Acceptance Criteria:~~ | ~~Procedures for establishing acceptable bolting patterns for the baffle-to-former bolts in Westinghouse-designed plants have been established in [13] and [14]. This methodology has been reviewed and accepted by the NRC is a Safety Evaluation in 1998 (TAC No. MA1152). The same methodology should be applied to the two operating CE plants with bolted core shrouds.~~<br>~~1. Observed pattern of unfailed bolts meets pre-defined acceptance criteria or is analyzed in real time for acceptability.~~<br>~~2. Less than 50% of initial margin consumed.~~<br>~~$N_f < (N - N_{req})/2$ (See below)~~<br>~~The margin is defined in terms of the number of intact bolts beyond the number required for the acceptable bolting pattern. The margin, M, at any time is simply:~~<br>~~$M = N - N_{req} - N_f$~~<br>~~Where~~<br>~~N = total number of shroud former bolts~~<br>~~$N_{req}$ = number of shroud former bolts in acceptable pattern~~<br>~~$N_f$ = number of failed bolts.~~<br>~~Assuming that there are no failed bolts at the beginning of life, the initial margin is simply: $(N - N_{req})$. $N_{req}$ is determined via an Acceptable Bolting Analysis. For operation through the next 10-year interval, require that no more than 50% of initial margin be consumed at the time of the inspection. Require that no more than 50% of initial margin be consumed for any subsequent inspection as well.~~ |
| ~~Approach:~~ | ~~No generic effort required. Only one plant is affected.~~ |
| ~~Additional Action(s):~~ | ~~None~~ |



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C-4

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



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C-5

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



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C-6

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



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C-7

| | |
|---|---|
| ~~CE ID: 1.2~~ | ~~Core Shroud Assembly (Bolted)~~<br>~~Barrel-Shroud Bolts~~ |
| ~~Analysis:~~ | ~~The observed pattern of failed bolts must meet the pre-defined acceptable bolt pattern and have a reasonable margin to protect against additional failures during the inspection interval. The margin is defined in terms of the number of intact bolts beyond the number required for the acceptable bolting pattern. The margin, M, at any time is simply:~~<br>~~M = N - Nreq - Nf~~<br>~~Where~~<br>~~N = total number of barrel-former bolts~~<br>~~$N_{req}$ = number of barrel-former bolts in acceptable pattern~~<br>~~$N_f$ = number of failed bolts.~~<br>~~Assuming that there are no failed bolts at the beginning of life, the initial margin is simply: (N-$N_{req}$). $N_{req}$ is determined via an Acceptable Bolting Analysis. For operation through the next 10-year interval, require that no more than 50% of initial margin be consumed at the time of the inspection. Require that no more than 50% of initial margin be consumed for any subsequent inspection as well.~~ |
| ~~Acceptance Criteria:~~ | ~~Procedures for establishing acceptable bolting patterns or the barrel-to-former bolts in Westinghouse designed plants have been established in [14]. This methodology has been reviewed an accepted by the NRC in a Safety Evaluation in 1998 (TAC No. MA1152). The same methodology should be applied to the two operating CE plant with bolted core shrouds.~~<br>~~1. Observed pattern of unfailed bolts meets pre-defined acceptance criteria.~~ |
| ~~Approach:~~ | ~~No generic effort required. Only one plant is affected.~~ |
| ~~Additional Action(s):~~ | ~~None~~ |



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

**Core Shroud Plate-Former Plate Weld**

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

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

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

| | |
|---|---|
| **Failure** | |
| Failure Mechanism: | Cracking (IASCC) |
| Failure Effect: | 1. Damage to peripheral fuel assemblies.<br>2. Through-wall crack provides leak path through shroud. |
| Failure Criteria: | An existing flaw is unacceptable if the flaw length projected at the next inspection cycle is beyond the allowable crack length. .<br>No observable damage in corresponding sections of peripheral fuel assemblies. |
| **Methodology** | |
| Goal: | Demonstrate that the cracking mechanism will not result in growth beyond the allowable crack length over the planned inspection interval. |



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

**Core Shroud Plate-Former Plate Weld**

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

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

Note:

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

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



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C-33

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



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C-86

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

**Inspection**

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

**Inputs and Assumptions**

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

- The inspections identified in MRP-227 are intended to provide a sampling of potential locations of degradation. Under this approach, inspection of one side (surface) of the weld is intended to provide an adequate sampling for monitoring of fatigue.
- The change in resistance to fracture of the RV core barrel welds can be correlated to the accumulated fluence at each weld location. Welds that are subject to low fluence are considered to have a high degree of resistance to fracture. Correspondingly, those welds subject to high fluence have lower resistance to fracture.
- The prediction of crack growth is based on the stress intensity factor, K, calculated using linear elastic fracture mechanics. The rate of crack growth is dependent on the amount of neutron fluence that the weld is expected to accumulate over the licensed operating lifetime. Since there has been very limited experience of SCC initiated cracks in operating PWRs to date, growth rates developed for the prediction of SCC in BWRs are assumed to be appropriate for prediction of crack growth due to SCC in PWR reactor internals.
    - For weld locations subjected to fluence less than or equal to $5x10^{20}$ n/cm$^2$ (E>1MeV), the boiling water reactor (BWR) hydrogen water chemistry (HWC) crack growth equation specified in paragraph C-8520 of Appendix C of Section XI of the 2010 edition of the ASME Boiler and Pressure Vessel Code is appropriate. This crack growth rate model is consistent with the model in BWRVIP-14-A, which was previously reviewed and approved by the NRC.



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C-91

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|---|---|
| **CE-ID: 9** | **Lower Support Structure**<br>**Core Support Plate** |
| Category: | Primary Applicability: All plants with a core support plate |
| Degradation Effect: | See MRP-227 Table 4-2 |
| Expansion Link: | See MRP-227 Table 4-2 |
| Function: | Primary core support. Provides alignment of fuel assembly. |
| **Inspection** | |
| Method: | ~~If screening for fatigue cannot be satisfied by plant-specific evaluation, visual examination is required.~~ See MRP-227 Table 4-2. |
| Coverage: | See MRP-227 Table 4-2. |
| Observable Effect: | The specific relevant condition is a detectable, crack-like surface indication. |
| **Failure** | |
| Failure Mechanism: | Aging Management (IE), Cracking (fatigue) |
| Failure Effect: | 1. Loss of core support.<br>2. Difficulty in loading fuel due to misalignment. |
| Failure Criteria: | Deflection of the core support plate beyond the maximum permissible to maintain alignment of the fuel assemblies. |
| Failure Assumptions: | • The acceptance criteria methodology assumes that cracking due to fatigue is the active degradation mechanism and that all other components in the lower internals assembly are in a nominal, nondegraded condition. If additional flaws are found in the lower internals assembly, additional analysis of the as-found assembly is required to determine suitability for return to service.<br>• Core support plate analysis assumptions:<br>1. If the flaw location is not defined, assume crack initiates at the location in the plate with highest surface tensile stress.<br>2. Assume crack propagates through the full thickness of the plate through the ligament with highest stress to the adjacent hole with highest stress. |
| **Methodology** | |
| Goal: | Cracks in the core support plate are expected to grow from hole-to-hole within the plate. The goal is to demonstrate that the cracking, which might be present, does not cause sufficient stress in other ligaments to violate functional requirements. |
| Data Requirements: | 1. Operating loads (e.g., pressure, flow, and thermal for limiting conditions).<br>2. Functional requirements for the core support plate assembly. |



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C-93

| | | | |
|---|---|---|---|
| **CE-ID: 10** | **Upper Internals Assembly**<br>**Fuel Alignment Plate** | | |
| Category: | Primary | Applicability: | All plants with core shrouds assembled with full-height shroud plates |
| Degradation Effect: | See MRP-227 Table 4-2 | | |
| Expansion Link: | See MRP-227 Table 4-2 | | |
| Function: | Provide fuel assembly alignment and support. Direct flow into upper internals. | | |
| **Inspection** | | | |
| Method: | ~~If fatigue life cannot be demonstrated by evaluation, visual examination is required.~~ See MRP-227 Table 4-2. | | |
| Coverage: | See MRP-227 Table 4-2. | | |
| Observable Effect: | The specific relevant condition is a detectable crack-like surface indication. | | |
| **Failure** | | | |
| Failure Mechanism: | Aging Management (IE), Cracking (fatigue) | | |
| Failure Effect: | 1. Loss of core support.<br>2. Difficulty in loading fuel due to misalignment. | | |
| Failure Criteria: | Deflection of the fuel alignment plate beyond the maximum permissible to maintain alignment of the fuel assemblies. | | |
| Failure Assumptions: | • The acceptance criteria methodology assumes that cracking due to fatigue is the active degradation mechanism and that all other components in the upper internals assembly are in a nominal, nondegraded condition. If additional flaws are found in the upper internals assembly, additional analysis of the as-found assembly is required to determine suitability for return to service.<br>• Fuel alignment plate analysis assumptions:<br>1. If the flaw location is not defined, assume the crack initiates at the hole in the plate, with the highest surface tensile stress.<br>2. Assume the crack propagates through the full thickness of the plate through the ligament with highest stress to the adjacent hole. | | |
| **Methodology** | | | |
| Goal: | Cracks in the fuel alignment plate are expected to grow from hole-to-hole within the plate. The goal is to demonstrate that the cracking, which might be present, does not cause sufficient stress in other ligaments to violate functional requirements. | | |
| Data Requirements: | 1. Operating loads (e.g., pressure, flow, and thermal for limiting conditions).<br>2. Functional requirements for the upper core plate assembly. | | |



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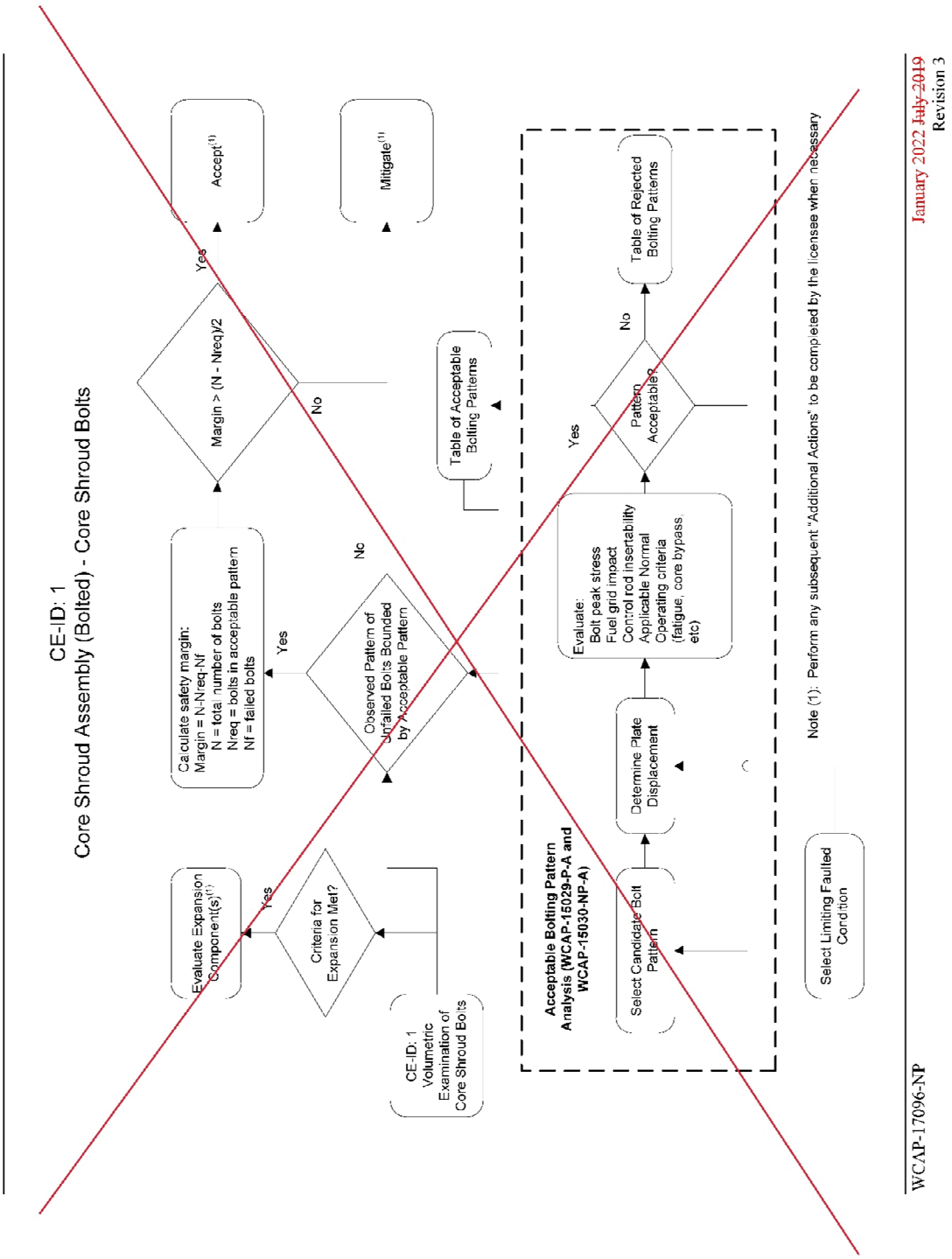
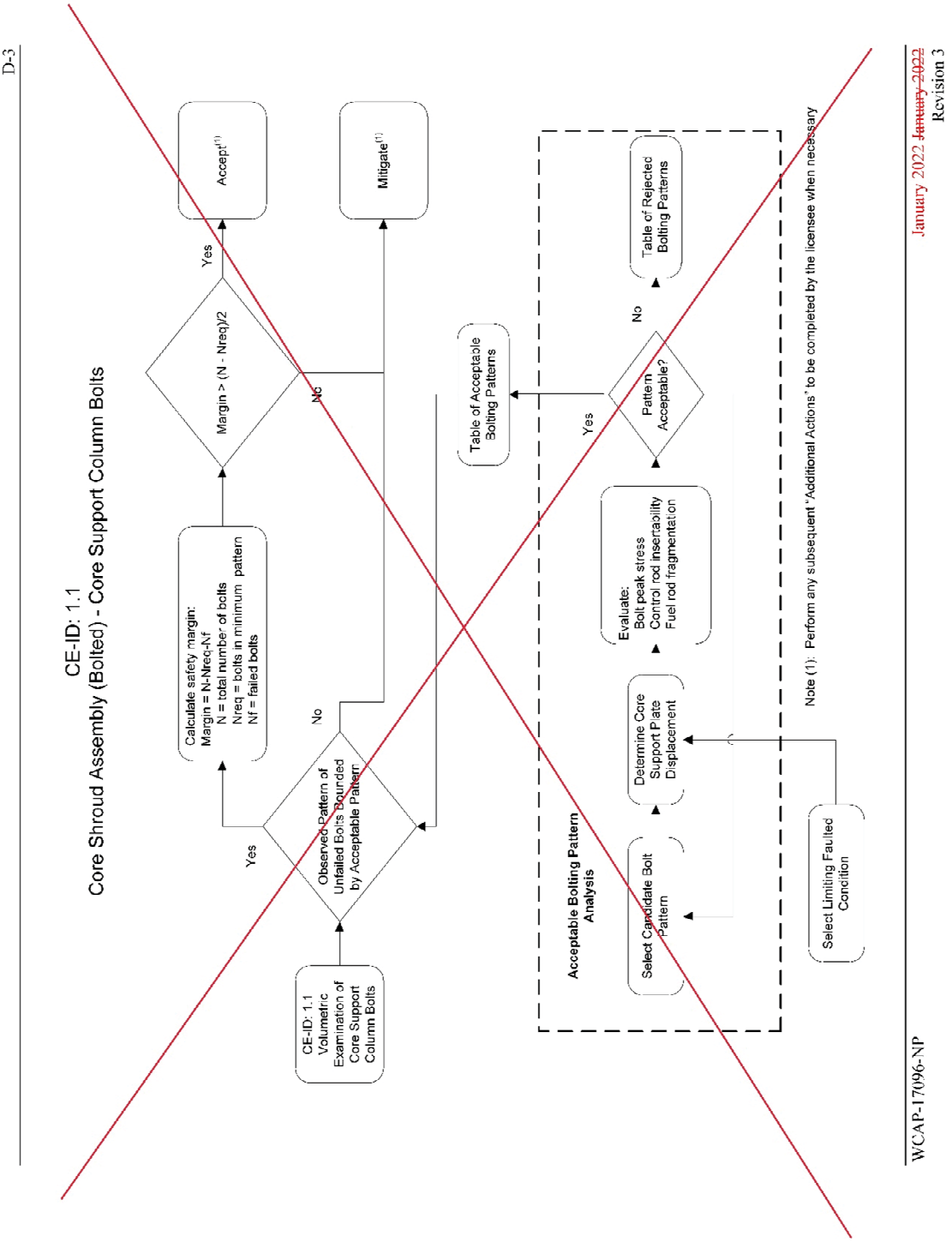


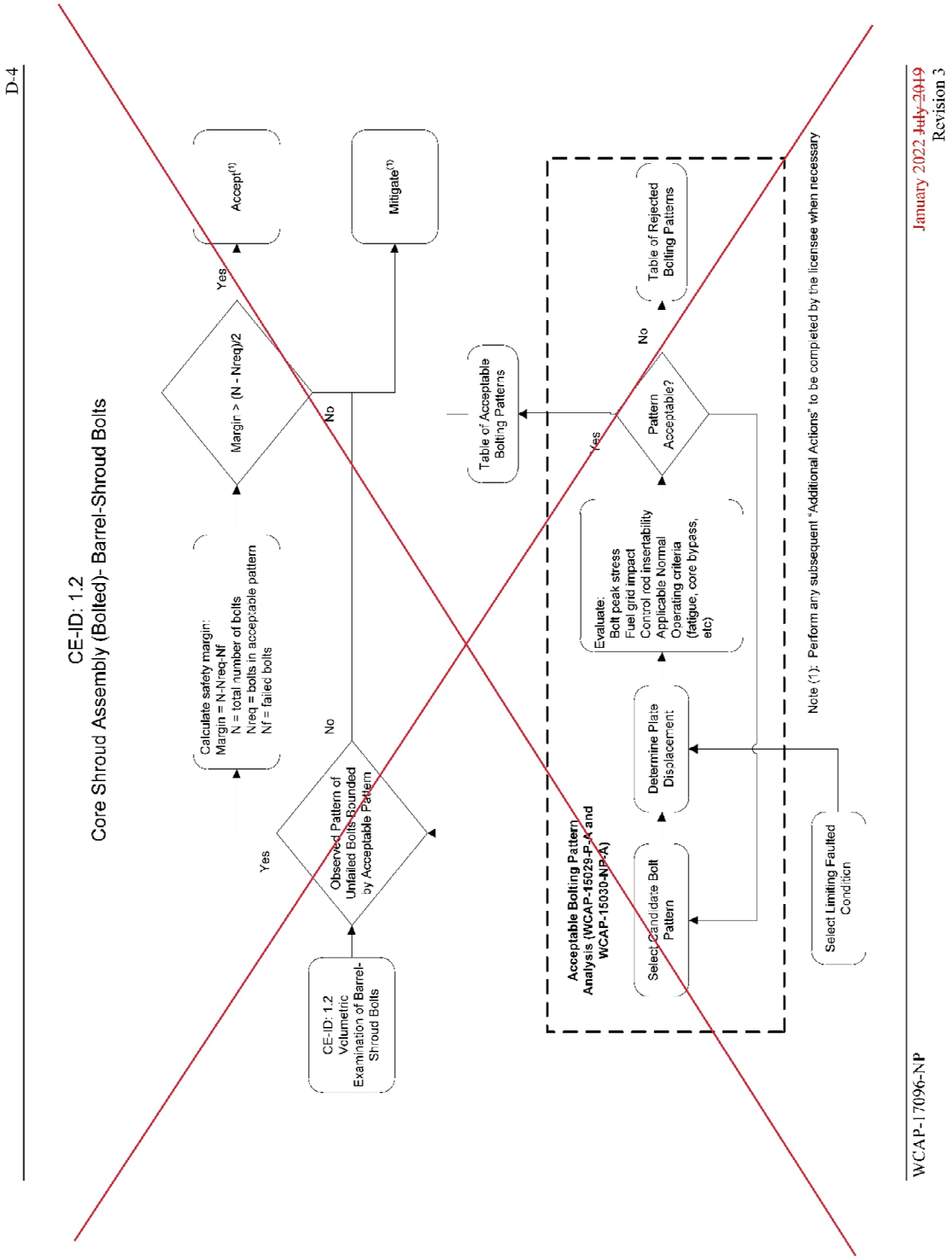



***This record was final approved on 10/4/2022 3:29:45 PM. (This statement was added by the PRIME system upon its validation)






***This record was final approved on 10/4/2022 3:29:45 PM. (This statement was added by the PRIME system upon its validation)






***This record was final approved on 10/4/2022 3:29:48 PM. (This statement was added by the PRIME system upon its validation)***

E-3

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

**Guide Plates (Cards)**

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

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

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

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

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

**Methodology**

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

Data Requirements:

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



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E-67

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



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E-70

**W-ID: 6** **Baffle-former Assembly**

**Baffle-former Bolts**

Margin Ratio (Typical Degradation only)

Less than 50% of the initial margin may be consumed. Therefore, the quantity of bolt failures upon re-inspection must be less than or equal to the number of bolt failures that has occurred to date.

$N_I < (N - N_{req})/2$

Where

N = total number of baffle-former bolts

$N_{req}$ = number of baffle-former bolts in acceptable pattern

$N_f$ = number of failed bolts

The margin (M) is

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

This degradation model may be applied at subsequent re-inspections, up to a 10-year re-inspection interval, when the "Typical" degradation category remains applicable.

Approach: ~~Generic work completed in previous PWROG program~~

Generic ABPAs that were previously performed should be confirmed to be applicable on a plant-specific basis.

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

Additional Action(s): 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.



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E-71

| | |
|---|---|
| **W-ID: 6.1** | **Core Barrel Assembly**<br>**Barrel-former Bolts** |
| Category: | Expansion Applicability: All plants |
| Degradation Effect: | See MRP-227 Table 4-6 |
| Primary Link: | See MRP-227 Table 4-6 and Interim Guidance MRP 2018-002 [48] |
| Function: | Maintain structural integrity of baffle-former-barrel structure. |
| **Inspection** | |
| Method: | Volumetric examination. See MRP-227 Table 4-6 |
| Coverage: | See MRP-227 Table 4-6 |
| Observable Effect: | Indication of flaw from UT examination |
| **Failure** | |
| Failure Mechanism: | Cracking<br>Loss of bolt pre-load due to irradiation induced stress relaxation may exacerbate fatigue issue in aging plants |
| Failure Effect: | Potential for flow induced vibration due to loss of bolting constraint.<br>Loss of structural stability |
| Failure Criteria: | Require an acceptable bolting pattern. |
| **Methodology** | |
| Goal: | Must demonstrate an acceptable bolting pattern. |
| Data Requirements: | Loads/displacements<br>Bolting patterns<br>Baffle-former-barrel design<br>Fast neutron (dpa) distribution in baffle-barrel region (if needed)<br>Projected bolt failure rate |
| Analysis: | Procedures for establishing acceptable bolting patterns for the barrel-to-former bolts have been established in WCAP-15029-P-A [13] / WCAP-15030-NP-A [14] as supplemented by PWROG-18034-P-A [64] and PWROG-18034-NP-A [65]. This methodology has been reviewed and accepted by the NRC in a Safety Evaluation issued in 1998 (TAC No. MA1152). |
| Acceptance Criteria: | 1. Observed pattern of unfailed bolts meets pre-defined acceptance criteria or is analyzed in real time.<br>2. Less than 50% of initial margin consumed.<br>$N_f < (N - N_{req})/2$ (See below)<br>The observed pattern of failed bolts must be shown to meet the conditions for being an acceptable bolt pattern per [13] / [14] as supplemented by [64] / [65] and have a reasonable margin to protect against additional failures during the inspection interval. The margin is defined in terms of the number of intact bolts |



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E-72

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

**Barrel-former Bolts**

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

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

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

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

Approach: Generic work completed in previous PWROG program.

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

Additional Action(s): None



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Westinghouse

To: James P. Molkenthin
cc:

Date: January 13, 2022

From: Materials and Aging Management
Ext: (412) 374-5559
Fax: (724) 940-8559

Our ref: LTR-AMLR-21-8, Rev. 0

Subject: **Responses to NRC Request for Additional Information 13 for WCAP-17096-NP, Rev. 3**

References:
1. U.S. Nuclear Regulatory Commission Letter, "U.S. NRC Request for Additional Information for Pressurized Water Reactor Owner's Group Topical Report WCAP-17096-NP, Revision 3, 'Reactor Internals Acceptance Criteria Methodology and Data Requirements'," September 21, 2020.
2. WCAP-17096-NP, Rev. 3, "Reactor Internals Acceptance Criteria Methodology and Data Requirements," July 29, 2019 (ADAMS Accession No. ML19218A179).

Attachment 1 of this letter provides the responses to the NRC Request for Additional Information (RAI) 13 (issued via Reference 1) on WCAP-17096, Rev. 3 (Reference 2). Attachment 2 contains the redline markups of the changes to WCAP-17096, Rev. 3 that were discussed in the RAI 13 response in Attachment 1.

Note that a draft version of this letter was provided to the NRC Staff on May 18, 2021. The draft RAI 13 responses were revised based on the discussion in the October 13, 2021 meeting with the NRC Staff. The redline markups of the changes to WCAP-17096, Rev. 3 to reflect the responses to RAI 13 were not revised from those contained in the draft version of this letter.

Framatome previously provided responses to RAI Numbers 2, 3, 9, and 12, and will provide a revised response to RAI 13(b).

If you have additional questions or require further information, please contact the undersigned.

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# ATTACHMENT 1

# Response to NRC Request for Additional Information RAI 13 on WCAP-17096-NP, Rev. 3



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**BACKGROUND:**

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

**NOTES:**

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

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

**Request for Additional Information (RAI) 13**

## NRC Request – Background

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

TR Section 2.3 (TR Pages 2-14 – 2-15) and TR Appendix E, W-ID Item 6, WEC Baffle-Former Bolts, and W-ID Item 9, WEC Thermal Shield Flexures.

Background:

TR Section 2.3 briefly addresses aging degradation OpE for WEC baffle-former bolts (BFB) and NEI 03-08 "Needed" BFB inspection guidance in MRP 2017-009 (incorporated by reference into MRP-227, Rev. 1-A). TR Appendix E, W-ID Item 6 incorporates the MRP 2017-009 guidance and further specifies the use of probabilistic modeling as a basis for evaluating BFB reinspection intervals. W-ID Item 6 states that the probabilistic model must predict both the proportion and spatial distribution (i.e., patterns) of failed bolts at certain time intervals. Acceptability of at least 95 percent of predictive model outputs is recommended as a basis for setting the reinspection interval. Acceptable bolting pattern analysis (APBA) for "as-found" inspection results and assessment of predictive pattern outputs is based on the earlier NRC-approved methodology of WCAP-15029-P-A, Rev. 1 (1999). The use of probabilistic modeling for predictive analysis of bolting degradation to establish reinspection criteria has never been formally approved by the NRC.

TR Appendix E, W-ID Item 9 provides evaluation methods for detected flaws in thermal shield flexures. These methods are generally consistent those in the NRC-approved Revision 2 of the WCAP (August 2016).

The NRC staff notes that due to the timing of its publication (July 2019), the TR did not have adequate time to address the following emergent degradation OpE and inspection guidance:

- Salem, Unit 1 BFB degradation that was found during spring 2019 outage inspections;
- Salem, Unit 1 thermal shield support block bolt (TSSBB) failures and crack-like indications in the thermal shield flexures, also found during the spring 2019 outage inspections;
- Issuance of WEC Technical Bulletin TB-19-5 (Reference 13), dated October 9, 2019, regarding thermal shield component degradation OpE at Salem, Unit 1 during the spring 2019 outage and associated inspection recommendations for WEC plants.



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On Sept. 18, 2019 the NRC staff and WEC conducted a noticed proprietary meeting to discuss the WEC probabilistic methods for predictive evaluations of BFB degradation. The WEC proprietary presentation materials (Reference 14) for non-proprietary version) provide information on the workings of the WEC methods. Staff is aware that at least one other vendor also provides probabilistic predictive analysis for degraded bolting patterns (BFB and other "bingo chart style" bolted assemblies), however other vendors have not provided the staff any information on the workings of their probabilistic methods for bolted assemblies.

*Salem, Unit 1 Baffle-Former Bolt Degradation:* Following its spring 2016 BFB replacement campaign, Salem, Unit 1 operated for two 1.5 year cycles (spring 2016 to spring 2019), and performed reinspection of the internals in spring 2019. Non-Proprietary Slide 41 of the September 2019 WEC presentation (Reference 14) provides a summary of the spring 2019 degradation findings for the Salem, Unit 1 BFBs, TSSBBs, and thermal shield flexures:

- Visual exams identified 31 degraded original BFBs;
- Subsequent UT exams identified 196 indications on original BFBs, 1 indication on 2016 replacement BFB, and 3 non-testable bolts;
- UT of TSSBBs identified 4 bolts with indications;
- Subsequent visual exam of TS flexures showed 2 flexures with confirmed cracks and 4 flexures with possible/inconclusive indications.

The September 2019 WEC presentation briefly addressed ongoing root cause investigations of the 2019 Salem, Unit 1 BFB reinspection results. At that time, the root cause of the 2019 unexpected BFB failures was identified as "indeterminate pending result of further studies" (Non-Proprietary Slide 46). Non-Proprietary Slide 43 addresses actions to investigate correlation between indications found on the thermal shield hardware and the unexpected BFB failures. Non-Proprietary Slide 45 offers a hypothesis that thermal shield (TS) flow-induced vibration (FIV) could relate to [baffle-former] bolt stresses (i.e., high cycle fatigue) via fluid coupling between the thermal shield assembly, core barrel, and baffle-former assemblies.

Non-Proprietary Slide 46 also indicates that WEC would continue to collaborate with industry on the "results of study investigating correlation between thermal shield and baffle bolts," and these communications would include corrective actions, industry communications, and "NRC communications." (Emphasis Added.)

## NRC Request – Part (a)

Issue:

The September 2019 WEC presentation provided some information on how the predictive evaluation outputs compare to BFB reinspection data. The staff needs additional information to verify that this method is suitable for setting reinspection intervals based on generating predictive outputs starting from an initial degraded pattern.

NRC Request:

(a) Considering the informational needs above, provide justification for the suitability of the predictive model based on how BFB reinspection results validate the predictive model outputs, where the initial condition is a degraded bolting pattern – i.e., a bingo chart of "reds" (cracked bolts based on UT) and "greens" (uncracked bolts based on UT).



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## Response to – Part (a):

The validity of the Westinghouse predictive model is demonstrated by considering the results from previous applications, which were used to predict both initial UT and re-inspection UT results. The September 2019 Westinghouse presentation on the development and application of the Westinghouse baffle-former bolt (BFB) predictive model (Rev. 0 [24] and Rev. 1 [14]) included a summary of the actual application of that particular model to individual plants. The predictive model has been applied to initial UT inspection for ten plants, which have included six 4-loop, two 3-loop, and two 2-loop Westinghouse NSSS plants. The predicted degradation from the model at the initial MRP-227 inspections were either in good agreement or conservative as compared to the UT inspection observations for all ten plants.

Comparisons of the re-inspection results for four 4-loop plants to the Westinghouse model predictions were included in the September 2019 Westinghouse presentation. Note that the re-inspection results are projected from the bingo chart of results from the initial MRP-227 inspection. The results for those re-inspections are shown on slides 50 and 51 of [24] (slides 51 and 52 of the Proprietary version of [24]) and slides 49 and 50 of [14] (slides 50 and 51 of the Proprietary version of [14]). These same results are shown graphically on slides 36 and 37 of [24] (slides 37 and 38 of the Proprietary version of [24]) and slides 35 and 36 of [14] (slides 36 and 37 of the Proprietary version of [14]). The UT re-inspection results were conservative (less than the 5$^{th}$ fractile for three quadrants and less than the 50$^{th}$ fractile for the fourth quadrant) for Plant E. The UT re-inspection results were in good agreement (less than the 95$^{th}$ fractile) for Plants D and F. Lastly, the UT re-inspection results for Plant C exceeded the results predicted by the Westinghouse model. However, the unexpected re-inspection results for Plant C were caused by factors not included in the scope of the BFB predictive model, which is discussed in the response to Part (c) of this RAI.

Based on this response, it is concluded that the previous applications of the predictive model versus initial and re-inspection of UT examination results demonstrate the validity of the model.

## NRC Request – Part (b)

Issue:

If other vendors use probabilistic methods for BFB that are different from those described in the September 2019 WEC presentation, the staff needs to assess whether the other methods are suitable for use in assessing BFB reinspection intervals.

NRC Request:

(b) Provide a list of the vendors that perform probabilistic evaluations for BFBs as part of W-ID Item 6. If any of these methods differ from those described in September 2019 WEC presentation, please make information available (via docketed response, audit, or other means) for the staff to assess the suitability of these methods.

## Response to – Part (b):

The process steps included in the predictive model methodology in W-ID: 6 provide a basic framework that can be applied by any capable vendor. Plant-specific data (UT inspection results, baffle-former assembly design information, and dose) are required inputs to the evaluation.



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Each application of the predictive model methodology in W-ID: 6 is required to be submitted to the NRC for information. This requirement is included as an "Additional Action" in the methodology and stipulates that the plant-specific evaluation be submitted to the NRC for information within one year of any degradation identified in MRP-227 inspections that exceeds the percentiles defined in MRP 2017-009 [17], or when evaluations are used to extend re-inspection intervals in MRP 2017-009. Therefore, the NRC would be informed of the vendor that performed the evaluation in cases when the results must be submitted for information.

In addition, the following text is included in the predictive model methodology, which explains that certain details of the predictive model methodology are proprietary and will not be included in the plant-specific submittal that was discussed above.

> "It is recognized that due to the proprietary nature of some inputs or models used in bolt probabilistic evaluations, portions of this content may not be included directly in the submittal. Per BWRVIP 2019-016, the submitter should have a plan for making the supporting materials available if needed for NRC review of the submittal."

Therefore, it is expected that the predictive model vendor will provide the proprietary details of the predictive model methodology to the NRC in support of the licensee. The slides in [14] from the Westinghouse presentation on the predictive model methodology provide compliance with this requirement.

Framatome also performs the predictive model evaluation of BFBs for Westinghouse NSSS plants. The methodology discussed in A.1.8 of [2] is applied to Westinghouse NSSS plants. Framatome provided further details of the methodology in A.1.8, and how this is applied to W-ID: 6 during the October 13, 2021 meeting with the NRC Staff [25].

## NRC Request – Part (c)

Issue:

In the Spring of 2019, the Salem, Unit 1 BFB failures were not expected given the 2016 bolt replacements and just three calendar years of operation between 2016 and 2019 inspections. The NRC staff is aware that a valid probabilistic evaluation should have predicted (within reasonable percentile bounds) the "as-found" results if all inputs to the evaluation were also valid. This points to potential error(s) in one or more of the following:

i. Evaluation of the spring 2016 UT data to identify cracked bolts (i.e., "reds" on the BFB bingo chart) versus non-cracked bolts ("greens" on the BFB bingo chart). Specifically, the NRC staff is concerned about the potential for UT to falsely identify "greens" on the bingo chart, where such greens may actually have a crack;
ii. Plant specific inputs, such as loading conditions, bolt material properties, component geometries, neutron dose, etc.; and/or
iii. One or more aspects of the methodology itself, such as modeling of crack initiation and growth, Weibull parameter(s), determination and use of ABPs, finite element modeling, IASCC effect based on neutron dose, etc.

NRC Request:



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(c) Provide a discussion of the root cause of the unexpected 2019 Salem, Unit 1 BFB failures and its resolution. This information should address resolution of potential contributors to the unexpected degradation, such as those listed above, and provide a basis for the staff to understand how the probabilistic methodology is suitable for assessing BFB reinspection intervals. Please address whether unexpected BFB degradation (i.e., degradation outside of percentile bounds for the predictive model outputs, such as for Salem, Unit 1 in spring 2019) remains a concern for current and future applications of this methodology. (Technical details beyond that included in the docketed correspondence can be made available for staff review via audit.)

## Response to – Part (c):

PSEG Nuclear LLC completed a revision to the root cause evaluation (RCE) for the unexpected high number of BFB failures at Salem Unit 1 [22]. Revision 2 of the RCE is included in INPO OE Report 455142. The scope of the RCE inherently addressed the potential contributors discussed above (i.e. items i, ii, and iii) when considering causal factors 1, 3, and 4 as discussed in [22]:

*"The scope of the Root Cause Evaluation is to investigate all of the following possible causal factors that led to the 272 baffle-former bolt replacement campaign during Salem 1R26:*

1. *The Westinghouse predictive evaluation will be examined to determine the cause of the analysis underestimating the total number of failed bolts after two Salem Unit 1 operating cycles.*

3. *Baffle-former bolt UT examinations during S1R24 were performed by AREVA (Framatome) and the S1R26 UT examinations were performed by Westinghouse. The UT processes and procedures for both companies will be examined to determine the capability/limitations and deltas.*

4. *Other potential failure modes of the significant clustering of the baffle-former bolts will be investigated."*

Causal Factor 1 was addressed by completing an investigation of the predictive model as discussed below [22]:

> *"First, during the S1R26 refueling outage, Westinghouse launched an investigation into the predictive model. This investigation . . . found no evidence which would question the validity of assumptions and/or underlying mechanics of the model for predicting IASSC initiation. Additionally, all inputs to the predictive model were confirmed to be valid including plant-specific inputs."*

In addition, PSEG Nuclear LLC requested that Westinghouse perform a sensitivity study to characterize the impact of the calibration of the predictive model from the full set of UT inspection results and bolt failure rates versus a calibration considering only those UT inspection results that were acquired in 2010 or later (i.e. after the MRP-228 inspection standard was first published in 2009). The full data set is based upon approximately 20 plants that performed 40 inspections, whereas the data set from 2010 or later included 17 plants and inspections. The results of this study are summarized below:

> *"This study . . . concluded that the use of the full UT inspection dataset was not the primary cause of the low predictions . . . and would have only provided a very small increase to the 95$^{th}$ fracture predictions (3% or less) had only UT inspection data obtained after 2010 been used."*



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Causal Factor 3 was addressed by reviewing the technical justification for the Westinghouse and AREVA (Framatome) UT examination process [22]. The section of the RCE that addressed Causal Factor 3 was redacted since that information is Proprietary to Westinghouse and Framatome. A summary of the results of Casual Factor 3 is provided below:

> PSE&G reviewed the technical justifications for the UT examination process used by Framatome and Westinghouse and determined that they were consistent with MRP-228 and ASME Section V, Article 14. Therefore, the RCE concluded that the requirements were met and the UT was performed satisfactorily.

Causal Factor 4 was addressed in the several investigative evaluations that were completed as part of the RCE [22]:

> *"As previously discussed, several investigative evaluations into the S1R26 thermal shield support degradation followed during the S1R26-S1R27 operating cycle… Most notably was a thermal shield vibration interaction study (see Corrective Action 70207205-0250), which focused on the potential relations between the thermal shield support degradation and the loads in other lower internals components. As discussed in . . . a dynamic finite element model (FEM) of the thermal shield, core barrel, and the baffle-former assembly was created and analyzed."*

The results of these studies indicated the following [22]:

> *"Based on the results from . . . it is clear that the significant baffle-former bolt degradation at Salem Unit 1 is unlikely to have caused the TSSBB failures identified during the 1R26 outage. However, on the contrary, the abnormal vibrations that result from the degradation of the thermal shield supports does have a direct effect on the stresses in the baffle-former assembly bolting. The most significant increase in baffle-former bolt and TSSBB stress was noted when simulating multiple nonfunctional thermal shield flexures."*

Additional information was provided by neutron noise (NN) data and laboratory testing of the TSSBB and BFBs [22]:

> *"As discussed in PSEG ACE 70215189, several additional investigations provided further evidence that nearly all of the RVI degradation discovered during S1R26 (i.e. TSSBB, flexures, and baffle-former bolts) could be connected to the abnormal vibrations of the thermal shield detected at Salem Unit 1 with NN monitoring."*

Based on the preceding evidence, the root cause in [22] is as follows:

> "*The root cause of the unexpected 232 failed baffle-former bolts during Salem S1R26 was the presence of unknown non-functional thermal shield flexures and the resulting abnormal vibration of the thermal shield, which caused an increase in baffle-former bolt stress and accelerated IASCC.*"

In addition, the RCE report provides the following conclusion for the underprediction of bolt failures by the BFB predictive model [22]:

> "*Because the Westinghouse baffle-former bolt predictive model (as well as the existing MRP-227 and WCAP-17096 aging management guidance) assumes design basis loads are acting on the internals, the introduction of additional stresses caused by this abnormal vibration were not*



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*accounted for when establishing the 3-cycle reinspection interval for baffle-former bolts following the S1R24 refueling outage. In other words, the presence of unknown non-functional thermal shield flexures and the resulting abnormal vibration of the thermal shield is now considered to be the root cause of the unexpectedly high rate of bolt degradation and the reason that the IASCC-based, probabilistic model was not able to accurately predict the degradation discovered at S1R26.*"

Based on this conclusion, the TR will be revised by adding the following text to Section 3 as shown on page 2 of Attachment 2:

"Note that following default assumptions have been applied in the development of the methodologies contained herein:

- The degradation of each component is assumed to be independent of the degradation of other components (i.e., no interaction), and the loading applied in the methodologies assumes the other components are functioning as designed."

In summary, the root cause evaluation concluded that degraded thermal shield flexures are the root cause of the unexpected BFB failures at Salem Unit 1 that were identified in the spring of 2019. The underprediction of bolt failures by the BFB predictive model was due to the increased bolt stress caused by additional vibration loads from the thermal shield, as a result of multiple degraded thermal shield flexures. The TR will be revised as discussed in the bullet above, to acknowledge that the degradation of each component is assumed to be independent of other components (i.e., no interaction).

Based on these conclusions, the unexpected degradation of BFBs is not a concern for future applications of the predictive model methodology provided that appropriate measures to manage the aging of thermal shield flexures per MRP-227, Rev. 1-A [6] are implemented, as well as the revised examination requirements in TB-19-5, Rev. 1 [21], and any future industry guidance on this topic.

## NRC Request – Part (d)

Issue:

W-ID Item 6 recommends a 95 percent probability of retaining an acceptable bolting pattern at reinspection based on evaluation of predictive model outputs against completed plant-specific APBAs. This is equivalent to a 5 percent probability of an unacceptable bolting pattern at reinspection. The staff does not consider a 5 percent probability of an unacceptable bolting pattern at reinspection to be an acceptable basis for setting the reinspection interval without justification that addresses its impact on plant risk.

NRC Request:

(d) Justify the 95 percent reliability criterion based on an assessment of the risk (e.g., core damage frequency) associated with having an unacceptable bolting pattern.

## Response to – Part (d):

Passive failures such as the failure of the BFBs are not typically modeled in plant Probabilistic Risk Assessments, since the failure probabilities associated with these passive failure mechanisms are typically orders of magnitude lower than other active failures. This is consistent with the requirements of the ANS/ASME PRA Standard [23] (see for example supporting requirement SY-A15).



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A conservative assessment of the risk associated with the BFB degradation was performed via a systematic review of thermal loads, flow induced vibrations, increased core bypass flow, baffle jetting, and RCCA insertion capability during normal operating conditions (including during startup and shutdown transients), upset conditions, and abnormal or faulted conditions. The assessment concluded that while an initiating event is not expected as a result of BFB degradation, pipe rupture events (i.e., LOCA events) can induce a pressure differential across the baffle plates which may be sufficient to generate fuel assembly grid crush.

A conservative core damage condition was defined for BFB degradation based on the following success criteria:

- The capability to insert all but one RCCA is not affected; and
- The Peak cladding temperature following the event is less than 2200°F, and
- The fuel cladding oxidation fraction following the event is within the limits of 10CFR50.46 for a LOCA; and
- The fuel rods retain sufficient structural integrity following the event to maintain a coolable geometry of the fuel, although cladding damage may occur due to baffle former bolt-related impacts.

Failure to meet any of these criteria for a given BFB-related event scenario were considered as core damage. It should be noted that this is a much more conservative success criterion than the core damage criteria used in PRAs as defined in the PRA Standard[1], which means that ultimately the CDF estimate through this simplified approach cannot be immediately combined with the CDF for the plant, but can be used as an indication of the bounding risk associated with BFB degradation.

The simplified risk assessment evaluates combinations of BFB patterns. Each combination is then assessed to see if the control rod insertion is impacted (which is not applicable to a large LOCA, which does not credit control rod insertion), if the 10CFR50.46 criteria for the peak cladding temperature and maximum cladding oxidation are met, and if the fuel rod structural integrity is maintained. Failure to meet any of these criteria is assumed to result in core damage. The frequency of failed scenarios for each of these conditions is accounted for in an estimate of the core damage. While dedicated analyses exist to assess the grid-crush status and thermal-hydraulic calculations also exists to address the 10CFR50.46 criteria, expert elicitation was used to assess the probabilities of the simplified event tree developed to address the risk for BFB degradation, and to group different configurations to existing analyzed cases. Uncertainties associated with the assumptions made via expert elicitation have been characterized and propagated in the estimates.

A detailed assessment of the seismic portion of the risk estimate would need to be plant-specific, however, a bounding hazard curve can be used. Developing a dedicated fragility analysis for the multiple different bolt failure patterns and conditions at the various plants is not a practical approach and bounding conditional probabilities of failures can be assumed, which are increasing with the seismic magnitude up to a guaranteed failure. Note that at high g levels, the increased risk due to the baffle bolt-related issues becomes negligible as the plant is likely already in core damage conditions due to multiple other failures.

[1] Core damage: uncovery and heatup of the reactor core to the point at which prolonged oxidation and severe fuel damage are anticipated and involving enough of the core, if released, to result in offsite public health effects [23].



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The simplified risk assessment supports that the combined risk estimated from BFB degradation is low per the RG 1.174 criterion for CDF.

In this perspective, the BFB probabilistic reliability analysis produces the array of bolting patterns predicted for the end of the re-inspection interval. These patterns are then evaluated based on the principles of the acceptable bolting pattern analysis (ABPA) methodology in WCAP-15029-P-A [18] as supplemented by PWROG-18034-P-A [19], and the results are compared to the 95 percent probability of maintaining an acceptable bolting pattern criterion.

The Westinghouse BFB predictive model includes conservatism which bounds the uncertainties applied to the various inputs and calculations of the model. Uncertainty is added to the inputs, including the neutron dose, bolt stress, and bolt geometry and is added to the model parameters, including model scale and shape parameters, dose threshold, and stress relaxation. These uncertainties are aggregated in the model calculations and provide a large spread in the predicted patterns (see the examples in slides 25 and 26 of [14]). Using a criterion of a 95 percent probability of an acceptable pattern from the patterns incorporating this model uncertainty, further adds significant conservatism to the re-inspection interval.

The primary objective of the WCAP-15029-P-A methodology is the evaluation of bolt stress, bolt strain when applicable (elastic-plastic analysis), and fuel grid impact during a Faulted condition event with degraded bolting patterns. The fuel grid impact evaluation ensures a coolable core geometry and control rod insertability, as discussed in NUREG-0800, Section 4.2, “Fuel System Design,” [20]. Numerous conservatisms are contained in an ABPA as discussed below:

- Seven criteria are evaluated in an ABPA (i.e. faulted bolt stress – both barrel-former bolts and BFBs, fuel grid impact, low and high cycle fatigue, momentum flux, and core by-pass flow). If any of the seven criteria is not met, then the bolting pattern is considered to be unacceptable; therefore, the process used to assess the criteria introduces conservatism.
- Each of the seven criteria consider “design” basis acceptance criteria; therefore, a safety margin requirement is included in the evaluation of each criterion.
- In general, the most limiting criterion for Faulted conditions in the ABPA methodology is the bolt stress (and bolt strain when applicable) rather than the fuel grid impact evaluation as applied to ensure a coolable core geometry and control rod insertability.
- The fuel grid impact evaluation considers a tiered evaluation approach that introduces conservatism. The initial approach compares the fuel grid loading to “grid crush” limits applicable to the fuel assembly design. If these limits are not met, then a more involved evaluation of “core coolability” is performed that would involve an assessment of peripheral and interior grid crush. The core coolability assessment of fuel assembly grid crush is performed pursuant to the requirements of 10 CFR 50.46.

Both the probabilistic reliability analysis and the ABPA methodology have inherent conservatism which represent an additional layer of conservatism on the simplified risk assessment discussed above, considering that an unacceptable bolting pattern does not directly translate to an accident scenario but represents the entry condition in the risk assessment discussed above.

In conclusion, the 95% reliability criterion provides an additional layer of conservatism to a scenario that is already judged to have low risk significance.



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## NRC Request – Part (e)

Issue:

The W-ID Item 6 analysis criteria state that, "the bolting pattern considers functional bolts to include those with satisfactory UT inspection results or replacement bolts, as well as any portion of untestable bolts determined to be acceptable per Step 2. [Emphasis Added] Otherwise, the un-inspectable bolts and those that show visual or UT indication of failure are treated as nonfunctional." Step 2 of this section includes a footnote stating, "untestable bolts should be reviewed on a plant-specific basis for determination if these should be credited as structural members when classifying clustering."

No specific guidance is provided on how a plant should evaluate untestable bolts to determine the extent to which they may be credited as functional structural members.

NRC Request:

(e) Provide justification for not including guidance on how a plant should evaluate untestable bolts to determine the extent they may be credited (wholly or partially) as functional structural members for W-ID Item 6.

## Response to – Part (e):

The examination requirements for BFBs, which are identified in Item W6 of Table 4-3 of MRP-227, Rev. 1-A [6], are "*100% of accessible bolts (Note 3),*" where Note 3 states: "*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.*" Therefore, MRP-227, Rev. 1-A allows the exclusion of a certain number of untestable bolts when performing the MRP-227 inspections of BFBs.

When evaluating reduced bolting patterns based on inspection results, in general, the most conservative approach should be applied, which is to consider all untestable bolts as failed. However, the use of sampling statistics was also determined to be an appropriate method to consider untestable bolts as discussed in the NRC Safety Evaluation for WCAP-17096, Rev. 2, which states:

> *"In response to RAI 32, Part 4, EPRI indicated that the uninspectable bolts would either be assumed to have failed, or sampling statistics would be used to determine the number of failures in the uninspected population. EPRI further indicated that the statistically determined number of failed bolts in the inspected population would be added to the number of failed bolts in the inspected population to determine the overall failed population $N_f$. The Staff finds this response acceptable because it provides a conservative means of estimated the total number of failed bolts."*

Similarly, the intent of the statement on untestable bolts as discussed in Part (e) of RAI 13 is to allow the use of sampling statistics in the W-ID: 6 methodology to credit untestable bolts as structural members when the untestable bolts are relevant to the determination of "clustering'" and the associated degradation category.

Note 1 of Step 2 in the TR will be revised to: "Sampling statistics may be applied on a plant-specific basis to credit untestable bolts as structural members when classifying clustering."

The revision to Note 1 of Step 2 in the Analysis section of W-ID: 6 is shown on page 3 of Attachment 2.



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## NRC Request – Part (f)

Issue:

The W-ID Item 6 acceptance criteria for the probabilistic model state that, "it is anticipated that this method will be used to demonstrate a 10 year re-inspection interval. Re-inspection intervals beyond 10 years may be justifiable in certain circumstances (e.g., acceptable bolting pattern relies on only replacement bolts and the plant is in, or has been converted to, an upflow configuration)."

The NRC staff identified that reinspection intervals need to be consistent with Table 4-3 of MRP-227, Rev. 1-A (Note 9 of that table) and NEI 03-08 "Needed" BFB inspection guidance in MRP 2017-009. Otherwise a deviation disposition per NEI 03-08 guidance would be required.

NRC Request:

(f) Provide justification for the noted inconsistencies between the language in W-ID Item 6 and Note 9 of Table 4-3 of MRP-227, Rev. 1-A and MRP 2017-009. Please cite and justify the use of applicable provisions from MRP 2017-009 (e.g., Paragraph C), as appropriate.

## Response to – Part (f):

It is acknowledged that note (b) in the Table 4-3 of MRP 2017-009 [17] states the following regarding the UT Re-Examination Period "is not to exceed 10-years,":

> *"A longer reinspection interval, not to exceed 10-years, may be justifiable by plant-specific evaluation based on plant-specific exam findings. This evaluation may include additional justification from plant modifications and/or improvements (for example: replacements of BFBs, conversion to up-flow, replacements of lower internals, etc.)."*

However, the statement was revised in the TR to recognize that unique circumstances may exist on a plant-specific basis such that re-inspection intervals beyond 10 years could be justified. However, it is acknowledged that this would result in a deviation to MRP-227 requirements that must be justified by the licensee.

Therefore, the statement in the last sentence in Step (c) of the probabilistic model in the TR will be revised to include the underlined text as identified below:

> *"Re-inspection intervals beyond 10 years may be justifiable in certain circumstances (e.g., acceptable bolting pattern relies on only replacement bolts and the plant is in, or has been converted to, an upflow configuration) based upon submittal of a deviation to MRP-227 requirements."*

The revision to the TR is shown on page 4 of Attachment 2.

Additionally, a typographical error was identified on page E-69 of the TR, in second process step (b), in the 2nd sentence and will be revised to:

"Justification shall be provided that an appropriate degradation rate…"

This revision is also shown on page 4 of Attachment 2.



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## NRC Request – Part (g)

Issue:

The W-ID Item 9 guidance for thermal shield flexures has not changed since the NRC-approved Revision 2 of the WCAP (August 2016), except for elimination of guidance on the use of neutron noise monitoring to monitor vibration and inclusion of the NRC submittal requirement. The spring 2019 OpE for TSSBB and thermal shield flexure degradation at Salem, Unit 1 and associated inspection guidance in WEC Technical Bulletin TB-19-5 show a need for an update to address these issues.

NRC Request:

(g) Provide justification for not including the inspection guidance from WEC Technical Bulletin TB-19-5 in W-ID Item 9.

## Response to – Part (g):

Westinghouse Technical Bulletin TB-19-5, Rev. 0 [13] was issued on October 9, 2019 in response to the degradation of the thermal shield flexures and support block cap screws that were identified at Salem Unit 1 in the spring of 2019. The TB discussed the inspection coverage, which includes both the inner and outer welds on the thermal shield flexures. This TB is referenced in MRP-227, Rev. 1-A. Additional inspections were performed at Salem Unit 1 in the fall of 2020 based on the inspection coverage recommendations in the TB. TB-19-5, Rev. 1 [21] was issued on February 22, 2021 and discussed the separation of the inner welds at five of the eight thermal shield flexure locations as Salem Unit 1. Revision 1 of TB-19-5 will be referenced in MRP-227, Rev. 2, rather than in W-ID Item 9 in WCAP-17096, because it is a more appropriate document, as the recommendations in MRP-227 are generally associated with the inspection of thermal shield flexures and support block cap screws (and not the acceptance criteria methodology in WCAP-17096).

It should be noted that the thermal shield flexure methodology in W-ID: 9 has been applied to develop acceptance criteria for MRP-227 inspections at multiple plants. This methodology postulates the existence of failed flexures, as well as the likelihood that one flexure failure would result in a cascading failure of additional flexures. Therefore, the methodology included in WCAP-17096-NP-A, Rev. 2 remains valid in light of the degradation identified at Salem Unit 1. No changes to the methodology in W-ID: 9 are required due to TB-19-5, Rev. 1.



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## References

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2. WCAP-17096-NP, Revision 3, "Reactor Internals Acceptance Criteria Methodology and Data Requirements," July 2019 (ADAMS Accession No. ML19218A179).
3. WCAP-17096-NP-A, "Revision Reactor Internals Acceptance Criteria Methodology and Data Requirements," August 2016 (ADAMS Accession No. ML16279A320).
4. NRC Correspondence Letter and Safety Evaluation from Mr. Kevin Hseuh (USNRC) to Ms. Anne Demma (EPRI), "Final Safety Evaluation of WCAP-17096-NP, Revision 2, 'Reactor Internals Acceptance Criteria Methodology and Data Requirements,'" May 3, 2016 (ADAMS Accession No. ML16061A243).
5. *Materials Reliability Program: Pressurized Water Reactor Internals Inspection and Evaluation Guidelines (MRP-227-A).* EPRI, Palo Alto, CA: 2011. 1022863. (ADAMS Accession Nos. ML12017A194, ML12017A196, ML12017A197, ML12017A191, ML12017A192, ML12017A195 and ML12017A199).
6. *Materials Reliability Program: Pressurized Water Reactor Internals Inspection and Evaluation Guidelines (MRP-227, Revision 1-A).* EPRI, Palo Alto, CA: 2019. 3002017168. (ADAMS Accession No. ML19339G350).
7. *Models of Irradiation-Assisted Stress Corrosion Cracking of Austenitic Stainless Steels in Light Water Reactor Environments, Volume: Disposition Curves Application*. EPRI, Palo Alto, CA, 2014. 3002003103.
8. EPRI Letter No. MRP 2019-023, "Transmittal of MRP-227-A-Related Interim Guidance Regarding PWR Core Barrel," September 3, 2019 (ADAMS Accession ML19249B102; includes enclosure of MRP 2019-009 [RAI Set Ref. 9 below]).
9. EPRI Letter No. MRP-2019-009, "Transmittal of NEI 03-08 "Good Practice" Interim Guidance Regarding MRP-227-A and MRP-227, Revision 1 Core Barrel and Core Support Barrel Inspection Requirements," July 17, 2019.
10. NRC Correspondence Letter and Safety Evaluation from Mr. Dennis Morey (USNRC) to Mr. Brian Burgos (EPRI), "Final Safety Evaluation for Electric Power Research Institute Topical Report MRP-227, Revision 1, 'Materials Reliability Program: Pressurized Water Reactor Internals Inspection and Evaluation Guideline,'"April 25, 2019 (ADAMS Accession No. ML19081A001).
11. PWROG-18034-NP, Revision 0, "Updates to the Methodology in WCAP-15030-NP-A, Rev. 0, 'Westinghouse Methodology for Evaluating the Acceptability of Baffle-Former-Barrel Bolting Distributions Under Faulted Load Conditions,'" October 2018 (ADAMS Accession No. ML18306A491).
12. Final Safety Evaluation by the Office of Nuclear Reactor Regulation, Topical Report PWROG-18034-P, Revision 0, "Updates to the Methodology in WCAP-15030-NP-A, Rev. 0, 'Westinghouse Methodology for Evaluating the Acceptability of Baffle-Former-Barrel Bolting Distributions Under Faulted Load Conditions,'" May 26, 2020 (ADAMS Accession No. ML20134M356).



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13. Westinghouse Technical Bulletin TB-19-5, “Westinghouse NSSS PWR Thermal Shield Degradation,” Westinghouse Non-Proprietary Class 3, October 9, 2019 (ADAMS Accession No. ML19302F228).
14. Westinghouse Presentation to the NRC Staff (Non-Proprietary Version), “Westinghouse Baffle-Former Bolt Predictive Methodology,” Josh McKinley and Matt Palamara, September 18, 2019 (ADAMS Accession No. ML19274F472).
15. U.S Nuclear Regulatory Commission Letter, “U.S. NRC Request for Additional Information for Pressurized Water Reactor Owner’s Group Topical Report, WCAP-17096-NP, Revision 3, ‘Reactor Internals Acceptance Criteria Methodology and Data Requirements’,” September 21, 2020.
16. EPRI Letter, MRP 2017-018, “MRP Demonstration Protocol to Document Ultrasonic Examination Procedures for PWR Internal Bolting,” February 11, 2020.
17. EPRI 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,” July 27, 2016. (ADAMS Accession No. ML16211A054).
18. WCAP-15029-P-A, Rev. 1, “Westinghouse Methodology for Evaluating the Acceptability of Baffle-Former-Barrel Bolting Distributions Under Faulted Load Conditions,” January 1999.
19. PWROG-18034-P-A, Rev. 0, “Updates to the Methodology in WCAP-15029-P-A, Rev. 1, ‘Westinghouse Methodology for Evaluating the Acceptability of Baffle-Former-Barrel Bolting Distributions Under Faulted Load Conditions’,” July 2020.
20. U.S. NRC Standard Review Plan 4.2, “Fuel System Design,” NUREG-0800, Rev. 3, March 2007.
21. Westinghouse Technical Bulletin, TB-19-5, Rev. 1, “Westinghouse NSSS PWR Thermal Shield Degradation,” Westinghouse Non-Proprietary Class 3, February 22, 2021.
22. PSE&G Report, 70207205, Revision 2, “Salem Unit 1 Unexpected High Number of Baffle-Former Bolt Failures,” March 17, 2021.
23. ASME/ANS RA-Sa-2009, Addenda to ASME/ANS RA-S–2008 Standard for Level 1/Large Early Release Frequency Probabilistic Risk Assessment for Nuclear Power Plant Applications.
24. Westinghouse Presentation to the NRC Staff (Non-Proprietary Version), “Westinghouse Baffle-Former Bolt Predictive Methodology,” September 18, 2019 (ADAMS Accession No. ML19242B346).
25. Framatome Document, ANP-3104, Rev. 3, Q1, Revision 1, “Framatome Responses to NRC WCAP-17096, Revision 3 RAIs,” November 2021.



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# ATTACHMENT 2

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

## Purpose

This attachment contains the redline markups of the changes to WCAP-17096-NP, Rev. 3 that are discussed in the responses to the U.S. Nuclear Regulatory Commission (NRC) Requests for Additional Information (RAI) Number 13.

These revisions will be included in the NRC approved version of WCAP-17096-NP, Rev. 3 that will be issued after the NRC Final Safety Evaluation is received.

## References

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



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# 3 EVALUATION PROCESS

The goal of this effort is to define the process to be used in the degradation evaluation procedures and the necessary data requirements to perform the evaluation of each Primary and Expansion component such that engineering organizations follow a consistent approach that is documented and approved. The guidelines and methodologies are listed in outline format and illustrated in a flowchart showing the evaluation and disposition options for the relevant condition.

Note that the following default assumptions have been applied in the development of the methodologies contained herein:

- The degradation of each component is assumed to be independent of the degradation of other components (i.e. no interaction), and the loading applied in the methodologies assumes the other components are functioning as designed.

Both Framatome and Westinghouse assembled teams of experts to develop recommendations. Although the procedural details and reporting format of these efforts are unique to the vendors, the basic process followed the same general steps:

1. Review MRP-227 degradation modes and inspection recommendations.
2. Determine component function.
3. List potential inspection outcomes and observable effects.
4. Identify potential failure mechanisms and effects.
5. Outline methodology to evaluate potential inspection observations.
6. Determine data requirements for inspection.
7. Consider potential for vendor-specific generic analysis of the components.
8. Identify additional actions.

The form and structure of the evaluation methodology is determined by the type of inspection recommended.

## 3.1 PHYSICAL MEASUREMENTS

Physical measurements are employed to characterize changes in component dimension. For example, measurements of internals hold down spring height are used to evaluate loss of core hold-down forces due to wear or stress relaxation. In this case, the required hold-down forces are a design requirement and generic or plant-specific acceptance criteria should be established prior to the inspection.

No specific action to establish general acceptance criteria was required under this task. The required methodologies simply note the existence of relevant design requirements for the affected components.

## 3.2 GENERAL CONDITION

There is a heavy reliance on general condition monitoring within the I&E recommendations. Although these recommendations refer to the VT-3 level visual inspections, the guidelines generally provide a description of the expected degradation. Under the VT-3 procedure, any observation of degradation is



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**W-ID: 6** **Baffle-former Assembly**

**Baffle-former Bolts**

2. The degradation is categorized as either "Typical" or "Accelerated" based upon plant design and the inspection results. The percentage of bolts with indications is based upon the fraction of the total quantity. This categorization considers the likelihood of additional bolt failures due to IASCC over a typical 10-year re-inspection interval. The degradation category for the observed and assumed failed bolts shall be determined using the following table:

| Plant Design | Inspection Results[2] | Degradation Category |
|---|---|---|
| Downflow | Less than 3% indications with no clustering[1] | Typical |
| | 3% or greater indications or clustering[1] | Accelerated |
| Upflow and Converted Upflow | Less than 5% indications with no clustering[1] | Typical |
| | 5% or greater indications or clustering[1] | Accelerated |

Notes:

1. Clustering is defined in MRP 2017-009 as three or more adjacent defective baffle-former bolts (BFBs) in the vertical or lateral directions on the same baffle plate, or more than 40% defective BFBs on the same baffle plate. Sampling statistics may be applied on a plant-specific basis to credit untestable bolts as structural members when classifying clustering.
2. These criteria are considered applicable to subsequent inspections until an appropriate degradation rate threshold is established through operating experience.

3. Additional margin to account for potential degradation through the next re-inspection interval must be provided. Plants that exhibit "Accelerated" degradation must apply the Probabilistic Model, but those with "Typical" degradation may apply either the Probabilistic Model or the Margin Ratio method as defined in the Acceptance Criteria.

Acceptance Criteria:

1. The analysis requirements as outlined above are applied to ensure structural integrity of the bolting pattern.
2. The degradation category is established consistent with industry guidance.
3. The evaluation must include margin by applying one of the following methods:



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E-69

**W-ID: 6** **Baffle-former Assembly**

**Baffle-former Bolts**

Probabilistic Model (Typical or Accelerated Degradation)

The evaluation applies the following process steps

a. Develop an empirical and/or mechanistic bolt degradation model from plant operating experience, plant-specific design information and test data. The model is calibrated to plant-specific bolt inspection results, or otherwise applies bolt inspection results from similar plants after accounting for relevant differences as appropriate.

b. Determine patterns of predicted failed bolts at the end of the re-inspection interval based upon the likelihood of failure. Justification shall be provided that an appropriate degradation rate has been applied. The evaluation is allowed to credit bolt replacements, or plant improvements such as upflow conversion.

c. A probabilistic reliability analysis is used to demonstrate reasonable assurance that the pattern of failed bolts will be acceptable at the end of the re-inspection interval. These techniques must consider both the total proportion of failed bolts and the spatial distribution, or pattern, of bolt failures. When predicting the total proportion of failed bolts, it is acceptable to consider results associated with the 95th fractile. For the potential spatial distribution of bolt failures, an evaluation showing acceptability of at least 95% of the predicted patterns at the end of the re-inspection interval can be used. It is anticipated that this method will be used to demonstrate a 10 year re-inspection interval. Re-inspection intervals beyond 10 years may be justifiable in certain circumstances (e.g., acceptable bolting pattern relies on only replacement bolts and the plant is in, or has been converted to, an upflow configuration) based upon submittal of a deviation to MRP-227 requirements.

EPRI has provided general guidelines for the minimum suggested content for probabilistic fracture mechanics submittals in letter BWRVIP 2019-016 [63]. At a high level based on this document, the minimum content for the bolt probabilistic reliability analysis can be summarized as:

- Description of model and underlying principals
- Required plant-specific or bounding inputs (bolt stress, dose, and inspection results)
  - Basis for uncertainties for each input
  - Treatment of input uncertainty in the evaluation
- Summary of benchmarking and convergence studies demonstrating the validity of the model and model results

It is recognized that due to the proprietary nature of some inputs or models used in bolt probabilistic evaluations, portions of this content may not be included directly in the submittal. Per BWRVIP 2019-016, the submitter should have a plan for making the supporting materials available if needed for NRC review of the submittal.



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framatome

# Framatome Responses to NRC WCAP-17096, Revision 3 RAIs

ANP-3104, Rev. 3,
Q1, Revision 0

## Technical Report

November 2021

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## Nomenclature

| Acronym | Definition |
|---|---|
| ASME | American Society of Mechanical Engineers |
| B&PV | Boiler & Pressure Vessel |
| B&W | Babcock and Wilcox |
| BB | Baffle-to-Baffle (Bolt) |
| BFB | Baffle-to-Former Bolt |
| CF | Core Barrel-to-Former (Bolt) |
| CFR | Code of Federal Regulations |
| IASCC | Irradiation-Assisted Stress Corrosion Cracking |
| LEFM | Linear Elastic Fracture Mechanics |
| MRP | Materials Reliability Program |
| NDE | Non-Destructive Examination |
| NP | Non-Proprietary |
| NRC | Nuclear Regulatory Commission |
| PWR | Pressurized Water Reactor |
| PWROG | Pressurized Water Reactor Owners Group |
| RAI | Request for Additional Information |
| RVI[1] | Reactor Vessel Internals |
| SE | Safety Evaluation |
| SSC | Systems, Structures, and Components |
| TR | Topical Report |
| VT-3 | Visual Testing |
| WEC | Westinghouse Electric Company |

[1] Used in quotations from the US NRC.



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## ABSTRACT

By letter dated July 31, 2019, the Pressurized Water Reactor (PWR) Owners Group (PWROG) submitted for U.S. Nuclear Regulatory Commission (US NRC) staff review WCAP-17096-NP, Revision 3, “Reactor Internals Acceptance Criteria Methodology and Data Requirements.” The US NRC has issued Requests for Additional Information (RAIs) on this submittal. This report provides the Framatome Inc. responses for RAIs 02, 03, 09, 12 and 13(b).



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# 1.0 INTRODUCTION AND SUMMARY

By letter dated July 31, 2019, the Pressurized Water Reactor (PWR) Owners Group (PWROG) submitted for U.S. Nuclear Regulatory Commission (US NRC) staff review WCAP-17096-NP, Revision 3, “Reactor Internals Acceptance Criteria Methodology and Data Requirements” (References 1 and 2). The US NRC has issued Requests for Additional Information (RAIs) on this submittal (Reference 3) and this report provides the responses to those RAIs assigned to Framatome Inc.

Upon receipt of the RAIs, the PWROG and Framatome Inc. reviewed the RAIs and determined who would respond to each RAI. The responses for RAIs 02, 03, 09, and 12 were assigned fully to Framatome Inc. The responses for RAIs 01, 04, and 13(b) were assigned to be shared between Framatome Inc. and Westinghouse. The remaining RAIs (05, 06, 07, 08, 10, 11, 13(a), 13(c), 13(d), 13(e), 13(f), 13(g), and 14) were assigned fully to Westinghouse. This document contains responses to RAIs 02, 03, 09, 12 and 13(b) only. The responses to RAIs 02, 03 and 12 propose changes to the WCAP-17096-NP, Revision 3 text.



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# 2.0 REQUESTS FOR ADDITIONAL INFORMATION

NRC RAIs 02, 03, 09, 12 and 13(b) are addressed in Section 2.1 through Section 2.5.

## 2.1 *RAI 02*

### 2.1.1 Statement of RAI 02

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

TR Section 2.2 (TR Page 2-5) and TR Appendix A, Babcock & Wilcox (B&W)-Inside Diameter (ID[2]) Items A.2.5 through A.2.9.

Background and Issue:

TR Section 2.2 and TR Appendix A, B&W-ID Items A.2.5 through A.2.9 include the following statement for resolving the NRC staff's Condition 1[3] for Group 1 B&W components.

> *"The licensee shall submit the plant-specific analysis for this component within one year after the inspection of the linked Primary item when the results trigger the expansion criteria in MRP-227."*

The TR applies this statement to the following Condition 1, Group 1 B&W "Expansion" category components:

a) core barrel cylinder and welds,

b) former plates,

c) core barrel-to-former (CF) bolts,

[2] This is assumed to be an erroneous acronym identification. A more appropriate acronym identification for "ID" in this case would be "identification".
[3] Note – Conditions and associating component Groupings are as specified in Section 4.0 in the NRC staff's Final SE (Reference 4) for Rev. 2 of WCAP-17096-NP.



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d) internal and external baffle-to-baffle (BB) bolts, and

e) locking devices for CF bolts and external BB bolts.

The staff notes that the above quoted TR statement for resolving Condition 1 for Group 1 B&W expansion items is not consistent with the guidance in Table 4-4 of MRP-227, Rev. 1-A since it does not include component replacement as an option. For these items (i.e., Item Nos. B9.1, B9.2, B10.1, B10.2, and B11.1 of Table 4-4 in MRP-227, Rev. 1-A), MRP-227, Rev. 1-A states: "*No examination requirements. Justify by evaluation or by replacement*." (Emphasis Added)

NRC Request:

Clarify the noted inconsistency in the above quoted TR statement for resolving Condition 1 for the five Group 1 B&W expansion components, which does not address the potential for component replacement as an alternative to plant-specific analytical evaluation, consistent with the guidance in Table 4-4 of MRP-227, Rev. 1-A.

### 2.1.2 Response to RAI 02

This wording was inadvertently not included in Section 2.2 and Appendix Sections A.2.5 through A.2.9 of the topical report (TR) (Reference 2). The updates to the text in these particular sections are addressed in Section 2.1.2.1 and Section 2.1.2.2.

#### 2.1.2.1 Resolution for TR Section 2.2

In Section 2.2 of the TR where the resolution to Condition 1 is addressed, the following text additions will be made, where text additions are noted in **bold font**:

> Group 1 RVI component items are addressed in WCAP-17096, Rev. 3 by the addition of the following statement in Appendix A, Sections A.2.5, A.2.6, A.2.7, A.2.8, and A.2.9: "The licensee shall submit the plant-specific analysis**, replacement schedule, or justification for some other alternative process** for this component item within one year after the inspection of the linked Primary item when the results trigger the expansion criteria in MRP-227."



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### 2.1.2.2 Resolution for TR Appendix A, Sections A.2.5-A.2.9

In Appendix A, Sections A.2.5 through A.2.9 of the TR under the 'Methodology and Data Requirements' heading, the following text additions will be made, where text additions are noted in **bold font**:

> The licensee shall submit the plant-specific analysis**, replacement schedule, or justification for some other alternative process** for this component item within one year after the inspection of the linked Primary item when the results trigger the expansion criteria in MRP-227.



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## 2.2 *RAI 03*

### 2.2.1 Statement of RAI 03

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

TR Section 2.2 (TR Page 2-5) and TR Appendix A, B&W-ID Item A.1.9.

Background and Issue:

Pages 13 and 45 of the staff's Final SE (Ref. 4[4]) for Rev. 2 of WCAP-17096-NP address the expansion of Condition 1 for Group 1 B&W Components to include the B&W baffle plates. B&W baffle plates are "Primary" category components addressed in Item B10 in Table 4-1 of MRP-227, Rev. 1-A and in TR Appendix A, B&W-ID Item A.1.9. The expansion of Condition 1 for Group 1 specifies that a linear elastic fracture mechanics (LEFM) analysis of the plant's baffle plates be submitted to the NRC staff within one year of their inspection if the inspection results indicate aging not meeting the acceptance criteria in Table 5-1 of MRP-227, Rev. 1-A. However, the TR's proposed resolution of Condition 1 for Group 1 B&W Components in TR Section 2.2 (TR Page 2-5) and TR Appendix A, B&W-ID Item A.1.9 does not address the expansion of this Condition to include the LEFM submittal needs for B&W baffle plates.

NRC Request:

Provide a justification for excluding the submittal of the baffle plate LEFM analysis to the NRC staff, consistent with the expansion of Condition 1 for Group 1 components in the Final SE for Rev. 2 of WCAP-17096-NP under the proposed resolution of Condition 1 for Group 1 B&W components in TR Section 2.2 (TR Page 2-5) and TR Appendix A, B&W-ID Item A.1.9.

[4] "Reference 4" is a direct quotation from the NRC RAI letter (Reference 3). The same reference herein is Reference 4.



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### 2.2.2 Response to RAI 03

This wording was inadvertently not included in Section 2.2 and Appendix Section A.1.9 of the TR (Reference 2). The updates to the text in these particular sections are addressed in Section 2.2.2.1 and Section 2.2.2.2.

#### 2.2.2.1 Resolution for TR Section 2.2

In Section 2.2 of the TR where the resolution to Condition 1 is addressed, the following text additions will be made following the proposed text for Group 1 (see Section 2.1.2.1), where text additions are noted in **bold font**:

> **Group 1 RVI component items are also addressed in WCAP-17096, Rev. 3 by the addition of the following statement in Appendix A, Section A.1.9: "The licensee is required to submit the LEFM analysis for the baffle plates under normal/upset condition loads, considering IASCC, within one year of the inspection of the Primary component item for NRC to determine whether review and approval are needed if the inspection results indicate aging not meeting the acceptance criteria in Table 5-1 of MRP-227."**

#### 2.2.2.2 Resolution for TR Appendix A, Section A.1.9

In Appendix A, Section A.1.9 of the TR under the 'Methodology and Data Requirements' heading, the following text additions will be made, where text additions are noted in **bold font**:

> **The licensee is required to submit the LEFM analysis for the baffle plates under normal/upset condition loads, considering IASCC, within one year of the inspection of the Primary component item for NRC to determine whether review and approval are needed if the inspection results indicate aging not meeting the acceptance criteria in Table 5-1 of MRP-227.**



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## 2.3 *RAI 09*

### 2.3.1 Statement of RAI 09

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

TR Appendix A, B&W-ID Items A.1.10, A.1.12, A.2.1, A.2.10, and A.2.12

Background and Issue:

TR Appendix A includes following statement for the above B&W items:

> "*A technical justification for removal* [of the applicable components associated with the B&W-ID items listed above] *may be necessary*."

It is not clear whether these statements are addressing a physical removal of the component through licensee performance of a justifiable SSC design modification or the removal of the component from either the "Primary" or "Expansion" inspection category in MRP-277[5], Rev. 1-A and recategorization as "No Additional Measures".

NRC Request:

Clarify whether the referenced statements are being made in reference to a potential design modification of the affected plant SSCs or a recategorization of the components, as defined in MRP-227, Rev. 1-A.

### 2.3.2 Response to RAI 09

The statement quoted above from Appendix A, Sections A.1.10, A.1.12, A.2.1, A.2.10, and A.2.12 is being made in reference to physical removal of the component item and/or weld (i.e., a design modification) in each case.

[5] This is believed to be a typographical error in the NRC quotation, which should be MRP-227.



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## 2.4 *RAI 12*

### 2.4.1 Statement of RAI 12

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

TR Appendix A, B&W-ID Items A.1.1, A.1.2, A.1.3, A.1.6, A.1.7, A.1.9, A.1.10, A.1.11, A.1.12, A.1.13, A.1.14, A.2.1, A.2.3, A.2.4, A.2.10, A.2.12, A.2.13

Background and Issue:

The B&W TR items listed above will receive either “Primary” VT-3 visual inspections or potential “Expansion” VT-3 visual inspections. The TR states that the general methodology to be used for the acceptance criteria for these components will be development of a non-destruction examination (NDE) inspection standard that contains examples of acceptable and unacceptable visual indications in the components. These statements do not address specific visual acceptance criteria that may be used to evaluate surface indications that are detected for these components.

NRC Request:

Provide the basis for the suitability of the VT-3 exam acceptance method without any information on specific acceptance standards for detected indications for these B&W components, or supplement these TR sections to include this information.

### 2.4.2 Response to RAI 12

The VT-3 exam acceptance criteria for TR Appendix A, B&W-ID Item A.1.1, is stated as follows:

*“The general methodology to be used for VT-3 (a general condition monitoring visual examination) acceptance criteria for these component items will be development of an NDE inspection standard that contains explanations of acceptable and unacceptable visual indications. Input information needed includes:*

- *Identification of the most likely signs and locations of wear that can be inspected*



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- *Identification of what visual examination wear indications are considered rejectable and would require additional dimensional examination and evaluation*
- *Identification of any additional examination results that are anticipated*
    - *Identify general acceptance criteria for the additional items expected to be in the VT-3 examination field of vision*

*Analytical efforts may be performed on a unit-specific basis due to the different initial interference. The NDE inspection standard could be developed generically."*

Similar statements are made in the TR for the VT-3 exam acceptance criteria for TR Appendix A, B&W-ID Items A.1.2, A.1.3, A.1.6, A.1.7, A.1.9, A.1.10, A.1.11, A.1.12, A.1.13, A.1.14, A.2.1, A.2.3, A.2.4, A.2.10, A.2.12, A.2.13.

It is understood that these statements from the TR do not fully define the acceptance criteria. Therefore, these statements shall be replaced with the following:

*"The VT-3 exam acceptance criteria are "no relevant conditions." Relevant conditions for the B&W Items of interest are defined in Section 5 of MRP-227-1A. These relevant conditions constitute discontinuities or imperfections (e.g., loose or missing parts, distortion, corrosion, wear, etc.), which are readily detectable with the VT-3 technique. Note that the VT-3 exam method meets the requirements of ASME B&PV Code Section XI, as specified and conditioned by 10 CFR 50.55a, and the MRP-228 Inspection Standard."*



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## 2.5 *RAI 13(b)*

### 2.5.1 Statement of RAI 13(b)

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

TR Appendix E, W-ID Item 6, WEC Baffle-Former Bolts

Issue:

If other vendors use probabilistic methods for BFB that are different from those described in the September 2019 WEC presentation, the staff needs to assess whether the other methods are suitable for use in assessing BFB reinspection intervals.

NRC Request:

Provide a list of the vendors that perform probabilistic evaluations for BFBs as part of W-ID Item 6. If any of these methods differ from those described in September 2019 WEC presentation, please make information available (via docketed response, audit, or other means) for the staff to assess the suitability of these methods.

### 2.5.2 Response to RAI 13(b)

In a meeting with the NRC Staff (Reference 5), Framatome provided a presentation of the methodology described in TR Section A.1.8 and how it is applied by Framatome to W-ID Item 6.



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## 3.0 REFERENCES

1. Letter OG-19-164, “PWR Owners Group ‘Transmittal of WCAP-17096-NP, Revision 3, “Reactor Internals Acceptance Criteria Methodology and Data Requirements,” PA-MSC-1567,” July 31, 2019, NRC Accession Number ML19218A178.
2. WCAP-17096-NP, Revision 3, “Reactor Internals Acceptance Criteria Methodology and Data Requirements,” July 2019, NRC Accession Number ML19218A179.
3. U.S Nuclear Regulatory Commission Letter, “U.S. NRC Request for Additional Information for Pressurized Water Reactor Owner’s Group Topical Report, WCAP-17096-NP, Revision 3, ‘Reactor Internals Acceptance Criteria Methodology and Data Requirements’,” September 21, 2020, NRC Accession Number ML20268B123.
4. WCAP-17096-NP-A, Revision 2, “Reactor Internals Acceptance Criteria Methodology and Data Requirements,” August 2016, NRC Accession Number ML16279A320.
5. NRC Closed Meeting with the Pressurized Water Reactor Owners Group (PWROG) on the Review of Topical Report WCAP-17096 Revision 3, "Reactor Internals Acceptance Criteria Methodology and Data Requirements,” October 13, 2021, NRC Accession Number ML21274A079.



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