May 3, 2016

Anne Demma
MRP Program Manager
Electric Power Research Institute
3420 Hillview Avenue
Palo Alto, CA 94304

SUBJECT: FINAL SAFETY EVALUATION OF WCAP-17096-NP, REVISION 2, “REACTOR INTERNALS ACCEPTANCE CRITERIA METHODOLOGY AND DATA REQUIREMENTS” (TAC NO. ME4200)

Dear Ms. Demma:

By letter dated May 19, 2010 (Agencywide Documents Access and Management System Accession No. ML101460156), the Electric Power Research Institute (EPRI) in cooperation with the Pressurized Water Reactor Owners Group (PWROG), submitted topical report (TR) WCAP-17096-NP, Revision 2, “Reactor Internals Acceptance Criteria Methodology and Data Requirements,” dated December 2009, for review by the U. S. Nuclear Regulatory Commission (NRC) staff. By letter dated November 12, 2015 (ADAMS Package Accession No. ML15265A128), an NRC draft safety evaluation (SE) was provided for your review and comment. By letter dated January 28, 2016 (ADAMS Accession No. ML16032A375), EPRI provided comments on the NRC draft SE. The EPRI comments and NRC staff responses are attached to the enclosed SE.

The NRC staff has found that WCAP-17096-NP is acceptable for referencing in licensing applications for nuclear power plants to the extent specified and under the limitations delineated in the TR and in the enclosed final SE. The final SE defines the basis for our acceptance of the TR.

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

In accordance with the guidance provided on the NRC website, we request that EPRI publish approved proprietary and non-proprietary versions of WCAP-17096-NP within three months of receipt of this letter. The approved versions shall incorporate this letter and the enclosed final SE after the title page. Also, they must contain historical review information, including NRC requests for additional information and your responses. The approved versions shall include an “-A” (designating approved) following the TR identification symbol.

As an alternative to including the RAIs and RAI responses behind the title page, if changes to the TRs provided to the NRC staff to support the resolution of RAI responses, and 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 RAIs and RAI responses can be included as an Appendix to the accepted version.
2. The RAIs and RAI responses can be captured in the form of a table (inserted after the final SE) which summarizes the changes as shown in the approved version of the TR. The table should reference the specific RAIs and RAI responses which resulted in any changes, as shown in the accepted version of the TR.

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

Sincerely,

/RA/

Kevin Hsueh, Chief
Licensing Processes Branch
Division of Policy and Rulemaking
Office of Nuclear Reactor Regulation

Project No. 669

Enclosure:
Final SE

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

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Enclosure:
Final SE

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FINAL SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

FOR TOPICAL REPORT WCAP-17096-NP, REVISION 2

"REACTOR INTERNALS ACCEPTANCE CRITERIA METHODOLOGY

AND DATA REQUIREMENTS"

PROJECT NO. 669

1.0 INTRODUCTION

The objective of the topical report (TR) process is, in part, to add value by improving the efficiency of other licensing processes, for example, the process for reviewing license amendment requests from commercial operating reactor licensees. The purpose of the U.S. Nuclear Regulatory Commission (NRC) TR program is to minimize industry and NRC time and effort by providing for a streamlined review and approval of a safety-related subject with subsequent referencing in licensing actions, rather than repeated reviews of the same subject.

A TR is a stand-alone report containing technical information or guidance about a nuclear power plant safety topic, which meets the criteria of a TR. A TR improves the efficiency of the licensing process by allowing the NRC staff to review a proposed methodology, design, operational requirements, or other safety-related subjects that will be used by multiple licensees, following approval, by referencing the approved TR. The TR provides the technical basis for a licensing action.

During the review of the Electric Power Research Institute's (EPRI's) TR WCAP-17096-NP, Revision 2, the NRC staff found that, in general, the TR meets the objectives of a TR and reinforces previously established NRC regulations and guidelines as noted within this safety evaluation (SE). The NRC has evaluated this TR against the criteria of Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50, and has determined that it does not represent a backfit. Specifically, NRC staff technical positions outlined in this SE are consistent with the aforementioned regulations and established staff positions, while providing more detailed discussion concerning the methodology and data required supporting reactor internals inspections. This SE endorses staff positions previously established through licensing actions and interactions with industry.

By letter dated May 19, 2010 (Ref. 1), as supplemented by letters dated June 14, 2012 (Ref. 2), January 3, 2013 (Ref. 3), August 5, 2013 (Ref. 4), January 16, 2014 (Ref. 5), April 10, 2014 (Ref. 6), July 8, 2014 (Ref. 7), October 31, 2014 (Ref. 8) and May, 5, 2015 (Ref. 9), EPRI, in cooperation with the Pressurized Water Reactor (PWR) Owner's Group (PWROG), submitted Westinghouse Non-Proprietary Class 3 TR WCAP-17096-NP, Revision 2 (Ref. 10, hereafter referred to as the TR), "Reactor Internals Acceptance Criteria Methodology and Data Requirements," for NRC review. The purpose of the TR is to provide methodologies for demonstrating PWR vessel internals integrity throughout the life of the plant, including the extended period authorized by license renewal in accordance with 10 CFR Part 54.

Enclosure

The objective of the 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 MRP-227-A report, "Materials Reliability Program (MRP) Reactor Internals Inspection and Evaluation (I&E) Guidelines" (Ref. 11) and;

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

The NRC staff reviewed the TR to determine whether it provides acceptable methodologies for developing acceptance criteria and evaluation procedures for the findings of the examinations of reactor vessel internals (RVI) components that will be conducted in accordance with the guidance of the MRP-227-A report. Several plants have developed, and more plants are currently developing, plant-specific RVI programs based on the guidance of the MRP-227-A report in order to meet license renewal commitments to implement the generic industry program for RVI.

2.0 REGULATORY EVALUATION

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

For RVI components, the Generic Aging Lessons Learned Report, Revision 1 (NUREG-1801), recommended for most RVI components that no further aging management review is necessary if the licensee provides a commitment in the final safety analysis report supplement to (1) participate in the industry programs for investigating and managing aging effects on reactor internals; (2) evaluate and implement the results of the industry programs as applicable to the reactor internals; and (3) upon completion of these programs, but not less than 24 months before entering the period of extended operation, submit an inspection plan for reactor internals to the NRC for review and approval.

The industry I&E guidelines for RVI components are contained in the MRP-227-A report. The SE for the MRP-227-A report was issued on December 16, 2011, and is incorporated in the MRP-227-A report. Plants that received renewed licenses that made a commitment to submit an inspection plan for reactor internals are submitting plant-specific inspection programs that reference the MRP-227-A report.

Although the MRP-227-A report supports the licensees in fulfilling the commitment to submit an inspection program, it does not contain detailed acceptance criteria and methodologies for evaluation of RVI components with relevant conditions for continued service. Instead, they are included in the TR.

In addition to the license renewal guidance, for plants licensed under 10 CFR Part 50 or 10 CFR Part 52, guidance to the NRC staff for review of RVI is provided in NUREG-0800, Standard Review Plan Section 3.9.5, "Reactor Pressure Vessel Internals," and Section 4.5.2, "Reactor Internal And Core Support Structure Materials." Relevant regulatory criteria include:

- General Design Criteria (GDC) 1 and 10 CFR 50.55a require that RVI be designed to quality standards commensurate with the importance of the safety functions performed.
- GDC 2 requires that RVI be designed to withstand the effects of natural phenomena such as earthquakes without loss of capability to perform safety functions.
- GDC 4 requires that RVI be designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operations, maintenance, testing, and postulated pipe ruptures, including loss-of-coolant accidents (LOCAs). Dynamic effects associated with postulated pipe ruptures may be excluded from the design basis when analyses demonstrate that the probability of fluid system piping rupture is extremely low under conditions consistent with the design basis for piping.
- GDC 10 requires that RVI be designed with appropriate margin to assure that specified acceptable fuel design limits are not exceeded during any condition of normal operation, including the effects of anticipated operational occurrences (AOOs).

3.0 TECHNICAL EVALUATION

3.1 EPRI's Evaluation

Section 1 of the TR provides the objectives for the TR. Section 2 provides the background on why this report was necessary. The TR contains detailed acceptance criteria and evaluation procedures for the RVI components, which were absent in the MRP-227-A report. This section of the TR provides an overview of the MRP-227-A report, including the inspection categories (Primary, Expansion, and Existing components), definition of a relevant condition and definition of acceptance criteria, and evaluation procedures.

Section 3 outlines the basic process used for developing the acceptance criteria and evaluation methodology for all the RVI components for all three nuclear steam supply system (NSSS) designs: Babcock & Wilcox (B&W), Combustion Engineering (CE), and Westinghouse Electric Company (Westinghouse). The general steps include identification of potential failure mechanisms and effects, outline of methodology to evaluate potential inspection observations, determination of data requirements for inspection, and consideration of potential for vendor-specific generic analysis. Additionally, this section describes the various types of inspections that will be performed in accordance with the MRP-227-A report recommendations: physical measurements, general condition monitoring, ultrasonic testing (UT), visual examination for cracking using VT-3 with enhanced visual EVT-1 or UT examination as a follow up, and VT-1 visual for void swelling. This section also discussed several components that are to be resolved by a time-limited aging analysis (TLAA) for fatigue (four CE plants), or other analysis (three expansion items in the B&W plants are considered inaccessible for inspection). However, in the response to follow-up RAI 3 dated May 5, 2015 (Ref. 9), EPRI clarified that guidance for TLAAs is not within the scope of WCAP-17096-NP.

Section 4 describes the format of the recommendations made in the TR. For each RVI component, the information from the MRP-227-A report Tables 4.1 and 4.4 was reproduced for the analysis of each Primary and Expansion component for the B&W plants. Likewise, the similar information from the MRP-227-A report Tables 4.2 and 4.5 was reproduced for the CE plants and Tables 4.3 and 4.6 for the Westinghouse plants. These tables contain the inspection recommendations for the Primary and Expansion components, including the category (Primary or Expansion), inspection method, coverage, and schedule. For each component, the report

also summarizes the function of the component based on the MRP-156 report, “Pressurized Water Reactor Issue Management Table, PWR-IMT Consequence of Failure.” The possible inspection outcomes were also summarized for each component under the headings “Observable Effects" and "Possible Outcomes” for the B&W plants, “Observable Effect” and “Failure Mechanism” for the CE plants, and “Failure Effect” and “Failure Criteria” for the Westinghouse plants. Methodology and data requirements are also included for each component. Methodology describes the techniques used to determine acceptability of the component such as a fracture mechanics technique, and data requirements could include inputs such as loads. The TR notes that implementation of the methodology and data requirements will generally require a combination of plant-specific and cooperative actions for specific vendor designs.

Section 5 summarizes the B&W plant results. The specific results are included in Appendix A and flowcharts of the evaluation process are included in Appendix B. Table 5-1 identifies components for which it may be feasible to develop generic acceptance criteria. Section 6 summarizes the CE and Westinghouse plant results. The specific results are included in Appendix C for the CE plants and Appendix E for the Westinghouse plants, and flowcharts of the evaluation process are found in Appendices D and F for the CE and Westinghouse plants, respectively. Table 6-1 identifies components for which it is feasible to develop generic acceptance criteria. Section 7 lists references.

3.2 NRC Staff Evaluation

3.2.1 General TR Sections

TR Sections 1 through 4 apply to all PWR plants, regardless of their NSSS design. Since EPRI’s evaluations under these sections are rather general and the quantitative guidelines (regarding mechanism, expansion link, examination method and frequency, and examination coverage) are presented in the TR Appendices A to F for each NSSS design, the NRC staff will discuss only the general issues that require justification or clarification in this subsection. Specific issues relevant to a specific category of RVI components under each NSSS plant design will be discussed later in Section 3.2.2 of this SE.

Since the TR is based on the MRP-227-A report, a general request for additional information (RAI) was issued regarding revision of the TR in accordance with the December 16, 2011, SE in the MRP-227-A report. EPRI’s response replied that the TR “will be revised to include the required changes made to MRP-227-A following issuance of the SE,” satisfying the NRC staff’s request. Therefore, the general RAI is resolved. It should be noted, however, that for most cases any limitations and conditions specified in this SE are considered as supplements (not replacements) to the limitations and conditions specified in the December 16, 2011, SE, contained in the MRP-227-A report. The exception will be discussed under the specific limitation and condition.

Section 2 of the TR states under “Evaluation Methodology” that the procedures and criteria to be used to evaluate relevant conditions include the engineering basis for repair, replacement, or mitigation options. Mitigation and repair are also included on the flow charts in Appendices D and F. However, it is not discussed in the evaluation sections for the individual components in Appendices C and E other than to indicate (for certain components) that if the acceptance criteria are not met, the components require mitigation. The NRC staff issued RAI 1, requesting that EPRI confirm detailed recommendations on mitigation and repair options are outside the scope of this TR. EPRI’s response states that, “The WCAP report will be revised to clearly state

that detailed mitigation and repair options are outside the scope of the document." Therefore, RAI 1 is resolved.

Section 3.7 of the TR states, "Three Expansion component items in the B&W designs have been designated for resolution by analysis." It is further stated in this section that, "Acceptance criteria for these three Expansion components are not included in this task." Since the analyses and their acceptance criteria for these three expansion components are not likely to be available soon, the NRC staff issued RAI 2, requesting that EPRI assess the impact of leaving these three Expansion components outside the TR's I&E plan. EPRI's response states, "Detailed analysis, replacement, or some other alternative process would be required if aging is detected in the linked [P]rimary component items." It further states, "the schedule to complete this work should be determined by each B&W licensee prior to the initial inspection of the [P]rimary component item." Providing just a schedule without any details for the proposed analysis or the alternative process prior to the initial inspection is not sufficient. Further, it is not logical to create a schedule without considering the initial inspection results of the Primary component item. Therefore, the NRC staff imposed a condition (Condition 1/Group 1) in this SE, requiring the licensee to submit the detailed analysis, replacement schedule, or justification for some other alternative process, within one year[1] 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 for the following RVI components (designated as Group 1):

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

The NRC staff determined that the "other alternative process" mentioned above shall include justification of operation in the degraded condition on a generic or plant-specific basis. The one year time shall not be used as justification for delaying submittals which are required to support continued operation under the current regulations. Condition 1 for Group 1 components will be repeated in Section 4 of this SE. Grouping of RVI components is intended to provide a better link between the evaluation presented in Section 3.2 of this SE and the conditions summarized in Section 4 of this SE. Group 1 RVI components are essentially those components listed in Applicant/Licensee Action Item 6 that are specified in the December 16, 2011, SE in MRP-227-A. 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. With this condition, RAI 2 is resolved.

Section 4.0 of the TR indicates that, for the B&W RVI components, the information regarding acceptance criteria methodology and data requirements contains a heading "What observations trigger examination into the Expansion category?" This information is not included for the CE and Westinghouse RVI components. The NRC staff issued RAI 3, requesting that the similar information be addressed in Appendices C and E for CE and Westinghouse RVI components. EPRI's response states that these expansion criteria are contained in Table 5-1 for B&W, Table 5-2 for CE, and Table 5-3 for Westinghouse in the MRP-227-A report. However, to have a consistent format in the TR for B&W, CE, and Westinghouse, the existing Appendix A

[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.

information for expansion criteria for B&W plants will be removed and information contained in the MRP-227-A report Table 5-1 will be incorporated by reference in Appendix A of the revised TR. This EPRI revision is acceptable because expansion criteria information is now consistent among Appendices A, C, and E. RAI 3 is resolved.

As mentioned in Section 3.1 of this SE, Sections 3 and 4 of the TR describe the basic process used for developing the acceptance criteria and evaluation methodology for managing degradation of RVI components and the recommendation format. Having resolved the RAIs discussed above, the NRC staff considers these high-level, general principles appropriate because based on the acceptance criteria and evaluation methodology, EPRI developed a framework, providing detailed guidance for managing degradation of each relevant RVI component of the three NSSS designs (Sections 5 and 6 of the TR).

Although the TR is meant to be self-contained from an application viewpoint, it is closely related to I&E guidelines of the MRP-227-A report that it supports. Therefore, the NRC staff's evaluation of top-level information such as binning the RVI components into Category A (insignificant effects), B (potentially moderately significant effects), and C (potentially significant effects) for each degradation mechanism; the MRP's categorization process (i.e., initial screening, failure modes, effects, and criticality analyses, or FMECA, and functionality analyses) for the development of an aging management program (AMP) for the RVI components; and MRP's development of I&E guidelines for the subject RVI components remains unchanged and can be found in the December 16, 2011, SE in the MRP-227-A report.

3.2.2 NSSS Design-Specific Information

3.2.2.1 <u>B&W Components</u>

This subsection evaluates the B&W plant components detailed in Appendix A of the TR. Since some component items which experienced similar degradation mechanisms are evaluated in the TR using the same methodology, the NRC staff places them into one category for ease of evaluation. For each category of the components, the NRC staff summarized the TR evaluation first, which is followed by the NRC staff's evaluation.

Before the NRC staff evaluates these categories of the RVI components, an issue generic to the B&W plants must first be discussed. For component items throughout Appendix A, it is mentioned frequently under "Methodology and Data Requirements," "perform analysis to…," or, "a … analysis is to be performed." Since each of these analyses will provide a basis for the acceptance criterion for the examination results and will justify the adequacy of the proposed examination frequency for an RVI component, the NRC staff, in RAI 4, requested EPRI to provide a schedule for completion of all the analyses which are mentioned in the TR as "to be performed" for the various RVI components. EPRI's response indicated that the plant-specific thermal and structural analyses have been completed for the upper core barrel (UCB) and lower core barrel (LCB) bolts for B&W units, but the schedule for completing the generic or plant-specific thermal and structural analyses for the flow distributor (FD), upper thermal shield (UTS), and lower thermal shield (LTS) bolts will be determined by each B&W licensee prior to initial inspection of the Primary component item. The schedule for plant-specific analyses on certain components will also be determined by each B&W licensee prior to initial inspection of the Primary component item. Consistent with the NRC staff's resolution of RAI 2, the NRC staff imposed a condition (Condition 1/Group 2) in this SE, requiring the licensee to submit, within

one year[1] of the inspection of the Primary or linked Primary component items, the analyses for specific RVI components (designated as Group 2) for NRC to determine whether review and approval are needed if 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 MRP-227-A:

- analysis for the top and bottom retaining rings of the vent valves
- thermal and structural analyses for the FD bolts
- thermal and structural analyses for the UTSand LTS bolts, with the analyses for UTS bolts including evaluation of joint stability of the thermal shield considering the flow-induced vibration (FIV)
- the analyses for Surveillance Specimen Holder Tube (SSHT) studs/nuts or bolts, and
- the analysis for Three Mile Island, Unit 1 (TMI-1) lower grid shock pad bolts.

Again, because of a lack of information about the above-mentioned analyses, Condition 1 for Group 2 components is needed for the NRC staff to determine whether review of these analyses is necessary. Condition 1 for Group 2 components will be repeated in Section 4 of this SE. With this condition, RAI 4 is resolved.

Core Clamping Items

The TR states that the core clamping load stabilizes and significantly restricts the rigid body pendulum motion of the core support and plenum assemblies. Adequate clamping load has been monitored indirectly by measuring the interference fit between pertinent reference surfaces. Loss of interference fit could occur due to wear of these surfaces. The acceptance criterion limits reduction in interference to no greater than 0.004 inch compared to the as-built data.

Since this acceptance criterion is based on engineering judgment, the NRC staff issued RAI 5, requesting EPRI to elaborate on this engineering judgment and demonstrate that after reduction of interference fit by 0.004 inch, there is still enough clamping load at operational conditions to prevent core movement due to the flow uplift loading and horizontal forces developed by various pump combinations. EPRI's response states, "Five of the seven B&W units have now been measured with no appreciable change in interference from the original as-built results. The original as-built results ranged from 0.004-inch interference to a 0.0004-inch clearance. Therefore, the Westinghouse Commercial Atomic Power (WCAP) report will be revised to include the foregoing sentences clarifying the original 0.004-inch criterion." The NRC staff accepts revision of the information regarding interference because operating experience (OE) supports the as-built interference of 0.004-inch to a clearance of 0.0004-inch and the original 0.004-inch criterion in the MRP-227-A report was based on the estimated permanent deformation in the O-ring mating surface, which is not considered in the interference measurement.

Regarding the VT-3 acceptance criteria for the relevant component items under Methodology and Data Requirement, EPRI's response to RAI 5 indicated that the schedule to complete a nondestructive examination (NDE) inspection standard, which contains explanations for

[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

acceptable visual indications, should be determined by each B&W licensee prior to the inspection. In a comment on the draft SE, EPRI further explained that, in practice, the technical justification for the VT-3 examinations was completed and ready prior to the inspection for the RVI component. Therefore, RAI 5 is resolved.

<u>Cast Austenitic Stainless Steel (CASS) Components</u>

Subject to thermal embrittlement (TE)

- Control Rod Guide Tube (CRGT) Spacer Castings

Subject to TE and irradiation embrittlement (IE)

- In-core Monitoring Instrumentation (IMI) Guide Tube Spiders and Welds

The RAI 6 and part of RAI 7 are related to core support shield (CSS) cast outlet nozzles, CSS vent valve discs, and CSS vent valve disc shafts, which are no longer in the inspection categories of MRP-227-A. The TR did not reflect this updated information before its publication; therefore, RAI 6 and this part of RAI 7 are no longer applicable. However, additional information regarding the function of vent valve assemblies in EPRI's response to RAI 6 is relevant and has been incorporated in the final version of the TR. The other updated information in MRP-227-A that the TR failed to note was that the CRGT spacer castings were re-categorized as Primary.

Unlike the CASS RVI components that are subject to TE, the IMI guide tube spiders and welds are also subject to IE. However, the proposed examination method/frequency and examination coverage are basically the same for both groups of the RVI components. The proposed VT-3 examination is acceptable because it is capable of detecting the possible examination outcomes (i.e., the IMI guide tube spider arms do not align with the lower fuel assembly support pad center bolt, or a spider arm is missing.) The proposed examination frequency is also acceptable, considering that the IMI guide tube spiders do not have a core support safety function and difficulty of insertion of an IMI caused by misalignment is not an issue during operation. Applicant/Licensee Action Item 7 that is specified in the December 16, 2011, SE in MRP-227-A requires plant-specific functionality analyses for B&W IMI guide tube assembly spiders and CRGT spacer castings be included as part of the submittal to apply MRP-227-A. The NRC staff imposed a condition (Condition 1/Group 3) in this SE, requiring the licensees update the plant-specific functionality analyses considering inspection findings and submit, within one year[1] of the inspection of IMI guide tube assembly spiders or CRGT spacer castings, 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 and not bounded by the previous analyses. Condition 1 for Group 3 components is needed for the same reason as Condition 1 for Group 2 components.

<u>Type 15-5 Precipitation-Hardenable (PH) Materials</u>

- CSS Vent Valve Retaining Rings

Although CSS vent valve retaining rings are made of Type 15-5 PH materials, their acceptance criteria, methodology, and data requirements are similar to the above CASS RVI components.

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[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.

For this component, the TR states, “Although there have been instances of failures of [PH] materials in other applications, there is no known history of OE identifying cracking in PWR reactor vessel internals applications.” The NRC staff issued RAI 7, requesting EPRI discuss the relevance of these “other applications” to the current situation in terms of their differences or similarities in the component stress and environment.

EPRI’s response states that OE with PH materials that have failed has been in applications involving high stresses. Since this response did not rule out the relevance of the failures of PH materials in other applications to the RPV internals of the same material, EPRI prudently included the PH CSS items in the RPV internals program for examinations. This is appropriate, therefore, RAI 7 is resolved.

<u>Alloy A-286 and Alloy X-750 Bolts and Locking Devices Subject to Intergranular Stress Corrosion (IGSCC)</u>

- UCB Bolts and Locking Devices
- LCB Bolts and Locking Devices
- UTS Bolts and Locking Devices
- SSHT Studs/Nuts or Bolts and Locking Devices
- Lower grid shock pad bolts and locking devices [(TMI-1)]
- LTS studs/nuts or bolts and their locking devices
- FD bolts and their locking devices

The TR specified the same effect (aging mechanism), the same examination coverage, but slightly different examination method/frequency for these items.

It is stated in the TR for the first item, “A change of no more than 20 % in stiffness when subjected to loss-of-coolant accident (LOCA) loads is acceptable.” The NRC staff issued RAI 8, requesting EPRI provide the basis for establishment of the acceptance criterion of less than 20 percent change in overall UCB stiffness and the confirmation whether UCB dynamics was considered in determining this criterion. The TR states, “An evaluation of joint stability (or, openness) is also to be performed.” RAI 8 requested EPRI define joint stability and openness in terms of parameters that can be calculated or measured. The TR also states the need to “incorporate the effect of future UCB bolt failure into the operability evaluation and re-inspection requirement.” RAI 8 requested EPRI to describe the methodology for establishing operability, including how operability is related to the overall core barrel stiffness, joint stability, and openness.

EPRI’s response revealed that AREVA NP Inc. has developed plant-specific analytical models and has performed analyses for UCB and LCB bolts using the criteria developed based on hypothetical failure patterns in these bolts. Since the horizontal dynamic response is frequency dependent, the number of core barrel bolts is considered acceptable if the UCB joint does not open, and thus does not change the response of the core barrel due to dynamic loadings. In case the joint opens and causes a fundamental structure change, then the overall stiffness of the core barrel will be limited to no more than 20 percent change so that the frequency change is under 10 percent, which is judged to be within the accuracy of the dynamic modeling. The response is acceptable because it has provided additional information on the analytical models requested by the NRC staff, including that accepting the change of 20 percent in UCB stiffness. The RAI response also defines joint stability (or openness) in the structural analysis in accordance with the “tight joint” criterion in the American Society of Mechanical Engineers

(ASME) *Boiler & Pressure Vessel Code* (Code) Section III, NG3232.1(e) requirement (i.e., the stress due to preload shall be greater than that due to primary and secondary membrane stress excluding preload). The response appropriately clarified the definition for a tight joint.

Regarding operability, EPRI's response states that an evaluation of OE with each of the high-strength bolt locations in B&W units will be part of the operability evaluation. Since information such as stress, material, time of service, number of failed bolts, etc. will be used to predict a conservative value for future bolt failures and to justify operability until the next UT examination, the response is acceptable.

Regarding OE and the proposed inspection frequency, EPRI's response (Ref. 2) states that most recent inspections have not identified any significant new failures except in one case where up to five LCB bolts were conservatively denoted as "failed" due to unusual UT responses. This OE raises no additional concern because it is bounded by the existing analysis based on hypothetical failure patterns in these bolts. The response further states, "Once the MRP-227-A inspections of these bolt locations are completed, the aggregate results are planned to be evaluated and either a revised inspection frequency will be proposed or the 10 year inspection frequency will be confirmed as appropriate by each B&W licensee prior to the follow-on inspection of the high-strength bolting component items." This is acceptable because either a revised inspection frequency considering future inspection results will be proposed or the 10-year inspection frequency will be confirmed, consistent with the December 16, 2011, SE in MRP-227-A.

Regarding the time to trigger examination of Expansion components, EPRI's response states that the expansion criterion for 10 percent of failed UCB bolts is based on redundancy, engineering judgment based on flaw tolerance of the assembly, and no safety consequences with the expansion bolts. The NRC staff considers the expansion criterion appropriate largely because of redundancy and no safety consequences. Based on the evaluation of EPRI's response to the NRC staff concerns discussed above, the NRC staff concludes that RAI 8 is resolved.

For LCB bolts and locking devices, the information in the TR is very similar to that for UCB bolts. RAI 9 thus requested EPRI to discuss the applicability of EPRI's response to RAI 8 for UCB bolts and locking devices to LCB bolts and locking devices. EPRI's response indicates that except for the frequency change due to opening of the UCB joint, the response to RAI 8 applies to LCB bolts. For the frequency change due to opening of the UCB joint, EPRI's response explained that unlike the UCB joint, which is located near the support of the core support assembly cantilever, the LCB joint is located near the free bottom end of the core support assembly cantilever. Hence, the dynamic system load change due to a loose LCB joint is not significant. Based on this information and the same basis for resolution of RAI 8, RAI 9 is resolved.

Likewise, for the UTS bolts and their locking devices, the information in the TR is very similar to that for UCB bolts. The NRC staff issued RAI 13, requesting EPRI provide justification for not addressing the effect of changing the fundamental frequency of the thermal shield (or losing firm attachment to the core barrel at certain places) due to failed UTS bolts or their locking devices. EPRI's response states, "an evaluation of joint stability is planned to be performed as part of the methodology prescribed. This will include consideration of a FIV analysis." This meets the NRC staff's request for an analysis considering the modeling of a bolted joint. However, EPRI's response states, "The schedule to complete this work should be determined by each B&W licensee prior to the initial inspection of the primary component item." Consistent with the NRC

staff's resolution of RAI 2 and RAI 4, the NRC staff requires the licensee to submit an evaluation of joint stability of the thermal shield considering the FIV, within one year[1] of the inspection of the linked Primary component item, 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. This is addressed as a Group 2 item in Condition 1 (the third bullet) as discussed in Section 3.2.2.1 of this SE. With this condition, RAI 13 is resolved.

RAI 14 requested the same information regarding the operability evaluation for the SSHT studs/nuts or bolts and locking devices. A similar response to RAI 13 was provided to RAI 14 in EPRI's response, and is acceptable on the same basis. For the plant-specific (only two plants have this feature) analyses for the SSHT studs/nuts or bolts, the NRC staff requires the licensee to submit, within one year of the initial inspection of the linked Primary component item, 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. This is addressed as a Group 2 item in Condition 1 (the fourth bullet) as discussed in Section 3.2.2.1 of this SE. With this condition, RAI 14 is resolved.

For the lower grid shock pad bolts and locking devices made of Alloy X-750 (applicable to TMI-1 only), it is stated in the TR that their function is to fasten the shock pads to the lower grid assembly to allow the pads to carry accidental core drop loads. The bolts do not carry the core drop load. The NRC staff issued RAI 19, requesting EPRI to substantiate the above statement by providing a sketch of the pad and the bolt, which shows the direction of the core drop load and the location of the impact. RAI 19 further requested EPRI provide detailed load distribution for the case when the load can be carried without the shock pads in place. EPRI's response provided schematics clearly showing how reactor vessel guide lugs and shock pads interact in case of core drop, meeting the NRC staff's request. Regarding the core drop for the postulated accident condition and its final position in the post-accident condition, EPRI's response further states, "The schedule to complete this work should be determined by the licensee prior to the initial inspection of the primary component item." The NRC staff determined that providing just a schedule prior to the initial inspection is not sufficient and the TMI-1-specific analysis shall be submitted, within one year[1] of the initial inspection of the linked Primary component item, 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. This condition is listed in Group 2 of Condition 1 (the last bullet) as discussed in Section 3.2.2.1 of this SE. With this condition, RAI 19 is resolved.

For the FD bolts and their locking devices, which are now a Primary component item in MRP-227-A, RAI 20 requested EPRI to explain how the results and conclusions from an existing generic FD bolt stress analysis (for all units except TMI-1) were utilized in preparing the acceptance criteria methodology and data requirements for this RVI component. EPRI's response states, "The FD bolt analysis will be performed either on a B&W generic basis in the PWROG or on a unit-specific basis by the licensee. The schedule to complete this work should be determined by the licensee prior to the initial inspection of the primary component item." Similar to resolution of RAI 19, the NRC staff imposed a condition requiring the thermal and structural analyses for the FD bolts be submitted, within one year of the inspection of the FD bolts, 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. This condition is

[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.

listed in Group 2 of Condition 1 (the second bullet) as discussed in Section 3.2.2.1 of this SE. With this condition, RAI 20 is resolved.

Core Barrel Assembly Bolts

- Baffle-to-former (BF) bolts
- BB bolts
- CF bolts
- Locking devices, including locking welds, for the external BB and CF bolts

For BF bolts, the TR states, "Past B&WOG work has determined the minimum number of bolts for safe shutdown, but not for operation." RAI 10 requested EPRI explain how this analysis was used to establish the criterion for conducting examination into the Expansion category, (i.e., 5 % (or 40) BF bolts with observed cracking or greater than 25 % of the bolts on a single former plate). EPRI's response states, "information contained in MRP 2008-036 (minutes of the Experts Panel meeting on expansion criteria for reactor internals I&E guidelines) contains the basis for the expansion criteria." The NRC staff reviewed the MRP 2008-036 meeting minutes and found that the expansion criteria were established appropriately because they are based on redundancy; past operability analyses supporting acceptable BF bolt rows in former elevations 3, 4, and 5; and functionality analysis supporting expected failures. Hence, RAI 10 is resolved.

For core BB bolts, the table column "Examination Coverage" indicated that an acceptable examination technique is not currently available for the internal BB bolts and that the external BB bolts are inaccessible. RAI 17 requested EPRI explain how the licensee can evaluate the degradation of the BB bolts using the FEA results developed by the MRP (i.e., an acceptable failure pattern or the number of failure bolts allowed) if the TR proposed no inspection for these BB bolts. RAI 18 requested EPRI address the same questions for the CF bolts. EPRI's response regarding RAI 17 indicates that analyses performed and to be performed for the core barrel assembly conservatively assumed that all BB bolts had failed. Therefore, no UT-inspection is required for the BB bolts. This is acceptable because the assumption of 100 percent BB bolt failures in the FEM analysis will bound any failure patterns in the BB bolts. RAI 17 is resolved. EPRI's response regarding RAI 18 indicates that the new FEM analysis includes various hypothetical failure patterns for the CF bolts and the analysis results will provide failure tolerance of the CF bolts in the core barrel assembly. Replacement is the option if it cannot be justified by evaluation. For the new FEM analysis, EPRI's response states, "The schedule to complete this work should be determined by each B&W licensee prior to the initial inspection of the primary component item." Since CF bolts are Condition 1/Group 1 components and the new FEM analysis for the CF bolts is simply a clarification of the "detailed analysis" mentioned before for Condition 1/Group 1 components, RAI 18 is resolved.

The locking devices and locking welds for the external BB and CF bolts are inaccessible. Since their failures are not a safety concern, but an indication of loose or broken external BB and CF bolts, the NRC staff accepts the proposed no inspection for these components.

Lower Grid Fuel Assembly Support Pad Items

The degradation mechanism of this RVI component is primarily irradiation embrittlement (IE). Various items are to be examined to identify separation or missing of any item, a dowel missing, or misalignment of the support pad. The proposed VT-3 is capable of detecting these general conditions, and, therefore, the NRC staff considers the proposed acceptance criteria methodology and data requirements acceptable for this RVI component.

Core Barrel Assembly BF Bolt and Internal BB Bolt Locking Devices Including Locking Welds

The degradation mechanisms of these RVI components are primarily irradiation-assisted stress corrosion cracking (IASCC), IE, and overload. Since distorted, loose, broken, or missing locking devices can be observed by the proposed VT-3 and the examination frequency of every 10 years is adequate for a component of no safety function, the NRC staff considers the proposed acceptance criteria methodology and data requirements acceptable for these RVI components.

Core Barrel Assembly Plates

- Baffle plates
- Core barrel cylinder (including welds)
- Former plates

The TR states that the baffle plates are subject to irradiation embrittlement and the expected crack opening displacement (COD) is determined for development of the inspection standard. The NRC staff issued RAI 11, requesting EPRI to discuss how the COD of a crack is going to be used in the inspection and in a flaw handbook with the calculated critical crack size. RAI 11 also requested EPRI list the situations that would make a VT-3 examination not possible for baffle plates. EPRI's response states that a flaw evaluation handbook is no longer considered for the baffle plates. Instead, a linear elastic fracture mechanics (LEFM) analysis is planned for a through-thickness flawed configuration with normal/upset condition loads, considering IASCC. Again, EPRI's response states, "The schedule to complete this work should be determined by each B&W licensee prior to the initial inspection of the primary component item." Similar to resolution of RAI 18, RAI 19 and RAI 20, the NRC staff imposed a condition (Condition 1/ Expanded Group 1) requiring the new LEFM analysis for the baffle plates be submitted, within one year[1] of the inspection of the baffle plates, 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. For better referencing, Group 1 components under Condition 1 will be expanded to include the baffle plates. With this condition, RAI 11 is resolved. It should be noted that the second part of RAI 11 regards potential justification of operation in the degraded condition on a generic or plant-specific basis. Since it is already addressed in resolution of RAI 2, it will not affect the resolution of RAI 11.

For the core barrel cylinder, since the examination coverage is designated as "Inaccessible" and no inspection is proposed, RAI 15 requested EPRI specify the condition that would trigger a need for justification by evaluation or by replacement and discuss, consistent with OE, the situations which would require disassembly of RVI and make inspection of the core barrel cylinder possible. Further, since the TR states, "The core barrel upper flange-to-core barrel

[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.

weld and upper heat-affected zone (HAZ) area is partially accessible and could potentially be VT-3 examined," RAI 15 requests EPRI provide guidelines for inspecting this partially accessible area and specify the level of degradation that can be determined from this partially accessible area such that a meaningful operability evaluation can be made for the core barrel cylinder to operate for at least one cycle.

EPRI's response states that cracking in two or more locations in the baffle plates shall require expansion into the core barrel cylinder with justification by evaluation or by replacement, satisfying the NRC staff request regarding when such a justification is needed. EPRI's response further indicated that an observation of extensive cracking on the top level of the upper former plate would require disassembly of the assembly (e.g., thermal shield removal) to allow inspection or replacement of the core barrel cylinder. The NRC staff considers the inspection strategy of not ruling out disassembly of the core barrel assembly prudent and appropriate. The last part of RAI 15 asked questions related to the core barrel upper flange-to-core barrel weld and upper HAZ area. EPRI's response states that it erroneously mentioned the core barrel upper flange-to-core barrel weld and upper HAZ area in the discussion of the core barrel cylinder. This area was classified as Category A (insignificant) in the MRP-227-A report and is irrelevant to the core barrel cylinder. This is acceptable, and the revision to the final TR will delete the irrelevant sentence. A proposed evaluation/analysis for the core barrel cylinder (one of the three inaccessible component items) was already discussed in Section 3.2.1 of this SE related to RAI 2, and the core barrel cylinder is a Group 1 item under Condition 1. Based on the above NRC staff evaluation and the condition, RAI 15 is resolved.

Except for minor differences, RAI 15 also applies to the former plates. RAI 16 requested EPRI to prepare similar responses for the former plates. The NRC staff reviewed EPRI's response and determined that the response to RAI 15 applies to RAI 16 also. Hence, the proposed treatment of the former plates is acceptable to the NRC staff for the same reason stated above. A proposed analysis for former plates was already discussed in Section 3.2.2.1 of this SE related to RAI 11, and the former plates were added to Group 1 under Condition 1. Based on the above NRC staff evaluation and the condition, RAI 16 is resolved.

<u>Alloy X-750 Upper or Lower Grid Assembly Components Subject to Primary Water SCC (PWSCC)</u>

- Lower grid assembly dowel-to-guide block welds
- Dowel-to-upper grid fuel assembly support pad welds or dowel-to-lower grid fuel assembly support pad welds

The TR states that the lower grid assembly dowel-to-guide block welds are potentially subject to PWSCC and possible outcomes of the examination of the dowel-to-upper grid fuel assembly support pad welds or dowel-to-lower grid fuel assembly support pad welds include one or several support pads are misaligned or missing. RAI 12 requested EPRI clarify whether loading of the fuel into the core will become a problem if this happened. RAI 12 also requested EPRI define the severity of the misalignment beyond which, loading of the fuel into the right position becomes a problem. EPRI's response states that the purpose of the VT-3 examination is not to measure the misalignment of each fuel assembly support pad. If misalignment causes problems for fuel loading, it would be identified during the fuel loading and the cause of the misalignment would be determined and addressed. This response clarified that these welds only "serve as loose part prevention devices" as stated in the TR. Separately, the TR mentioned, "acceptance criteria for the dowel-to-grid fuel assembly support welds will be development of a NDE inspection standard that contains examples of acceptable and

unacceptable dowel-to-grid fuel assembly pad welds visual indications." In a comment on the draft SE, EPRI further explained that, in practice, the technical justification for the VT-3 examinations was completed and ready prior to the inspection for the RVI component. Therefore, RAI 12 is resolved.

3.2.2.2 CE and Westinghouse Components

This SE section evaluates common CE and Westinghouse issues, which are those issues broadly applicable to many CE and Westinghouse-design RVI component-specific methodologies described in Appendix C and Appendix E of the TR, and also evaluates issues specific to groups of similar components.

3.2.2.2.1 Common CE and Westinghouse Issues

The NRC staff identified several issues that affected the analysis methodology or data requirements for several different components. These issues are addressed generically.

Changes Related to MRP-227-A

On June 22, 2011, the NRC staff issued its final SE related to MRP-227, Rev. 0, in which several conditions were identified requiring changes to be incorporated in the final approved version of MRP-227 (MRP-227-A). RAIs 45, 46, and 47 requested that EPRI modify the TR for consistency with Conditions 1-3 of the NRC staff's final SE on MRP-227, Rev. 0. These conditions generally require changes in the inspection categories for certain components. In its responses to RAIs 45, 46, and 47, EPRI indicated that the WCAP will be updated to reflect the changes required by the conditions from the MRP-227 SE.

In RAI 45, the staff asked EPRI to confirm that evaluation procedures for the Upper Internals Assembly – Upper Core Plate, Lower Internals Assembly - Lower Support Forging or Casting, Core Support Barrel Assembly - Upper Core Barrel Flange, and Lower Support Structure - Lower Core Support Beams, would be added to the TR for consistency with Condition 1 of the staff's final SE of MRP-227, Rev. 0, which required these components to be re-categorized from No Additional Measures to Expansion. EPRI's April 10, 2014, letter (Ref. 6) contained the evaluation procedures for these components. The new evaluation procedures are evaluated in Section 3.2.2.2 of this SE and are identified as:

- Upper Internals Assembly - Upper Core Plate (W-ID: 2.1)
- Lower Internals Assembly - Lower Support Forging or Casting (W-ID: 2.2)
- Core Support Barrel Assembly - Upper Core Barrel Flange (CE-ID: 6.3)
- Lower Support Structure - Lower Core Support Beams (CE-ID: 6.4)

RAI 45 is therefore resolved.

RAI 46 requested EPRI to verify that the Core Support Barrel Assembly - Lower Cylinder Girth Welds (CE) and the Core Barrel Assembly - Lower Core Barrel Flange Weld (W) and Core Barrel Assembly - Upper and Lower Core Barrel Cylinder Girth Welds (W) would be moved from Expansion to Primary inspection category in the final version on the TR, to reflect Condition 2 of the staff's final SE of MRP-227, Rev. 0. The staff verified the revised evaluation procedures for these components provided in EPRI's letters dated April 10, 2014 (for the CE components) and

July 8, 2014 (for the Westinghouse components) identify these components as Primary inspection category components. The new component designations are:

- CE-ID: 7 Core Support Barrel Assembly - Lower Cylinder Girth Welds
- W-ID: 4 Core Barrel Assembly - Upper and Lower Core Barrel Cylinder Girth Welds
- W-ID: 5 Core Barrel Assembly - Lower Core Barrel Flange Weld

Therefore, RAI 46 is resolved. Other changes to the evaluation procedures for these components were made in response to other RAIs, as discussed in Section 3.2.2.2.2.

RAI 47 requested that EPRI modify the TR to reflect the change from Expansion to Primary of the Lower Support Structure - Core Support Column Welds, which was required by Condition 3 of the staff's final SE of MRP-227, Rev. 0. The staff verified this change has been made in the revised evaluation procedure for the Lower Support Structure - Core Support Column Welds (CE-ID: 8) provided in EPRI's January 16, 2014, letter. RAI 47 is therefore resolved.

<u>Crack Growth Rate Models</u>

Neutron Fluence < $5x10^{20}$ n/cm^2

In response to RAI 29, Part 6, RAI 30, Parts 1a and 1b, RAI 34, Parts 1a and 1b, and RAI 35, Parts 2a and 2b, EPRI indicated the boiling water reactor (BWR) hydrogen water chemistry (HWC) CGRs would be used to model future crack growth in the subject components. EPRI referenced a 2007 Pressure Vessel and Piping (PVP) conference paper by Carter and Pathania (Ref. 13), as the basis for stress-corrosion cracking (SCC) CGR in a component with a fluence less than or equal to $5x10^{20}$ n/cm^2 at E>1.0 MeV. In its response to RAI 34, Part 1b, EPRI additionally referenced paragraph C-8520 of Appendix C of Section XI of the 2010 ASME Code for the CGR at fluences less than or equal to $5x10^{20}$ n/cm^2 at E>1.0 MeV (Ref. 14):

> $da/dt = (5.31x10^{-9})*K^{2.181}$ in/hr for fluence less than or equal to $5x10^{20}$ n/cm^2 at E>1.0 MeV

Equation 6-4 of BWRVIP-14-A, "BWR Vessel and Internals Project, Evaluation of Crack Growth in BWR Stainless Steel RPV Internals" (Ref. 15) provided a CGR of $da/dt = 0.983x10^{-8}\ K^{2.181}$ for BWR core shroud welds with fluence < $5x10^{20}$ n/cm^2, which is greater than EPRI's proposed CGR for fluence less than or equal to $5x10^{20}$ n/cm^2 by a factor of almost two. The NRC staff reviewed the technical basis for the proposed CGR and finds that the technical basis appears to justify both the proposed CGR and the BWRVIP-14-A CGR. Therefore, in follow-up RAI A, the NRC staff requested EPRI clarify this discrepancy. In response to RAI A, EPRI stated that the ASME Code, Section XI paragraph C-8520 CGR would be used and that it is consistent with the BWRVIP-14-A CGR. The NRC staff notes that both CGR equations are identical except that the ASME equation is expressed in da/dt rather than the natural logarithm of da/dt (ln(da/dt)). When the parameters from Table C-8520-1 (conductivity of 0.15 µS/cm, electrochemical corrosion potential (ECP) of -230 mV(SHE), and temperature of 530 °F) are directly used in this equation, the resulting CGR is $5.31x10^{-9}K^{2.181}$. The NRC staff verified that the BWRVIP-14-A CGR was derived using the same model, except that the environmental parameters for normal water chemistry (NWC) were used, and the CGR divided in half to account for the effects of HWC. Therefore, the BWRVIP-14-A CGR was derived from the same model, but used an indirect approximate approach. RAI A is thus resolved, since the NRC staff confirmed EPRI's proposed CGR was determined using an NRC staff-approved model.

Neutron Fluence ≥ $5x10^{20}$ and ≤ $3x10^{21}$ n/cm^2

For components with fluence ranging from $5x10^{20}$ n/cm^2 to $3x10^{21}$ n/cm^2 at E>1.0 MeV, in its response to RAI 30, EPRI referenced a 2009 conference paper by Pathania, Carter, Horn, and Andresen (Ref. 16). In response to Part 1b of RAI 30, EPRI confirmed that the BWR HWC CGR model is being used as recommended by MRP-227-A. In its response to RAI 34, Part 1b, EPRI additionally referenced BWRVIP-99, "BWR Vessel and Internals Project Crack Growth Rates in Irradiated Stainless Steels in BWR Internal Components" (Ref. 17) for the CGR in the fluence range of $5x10^{20}$ n/cm^2 to $3x10^{21}$ n/cm^2 at E>1.0 MeV. MRP-227-A, Section 6.0 provides the following equation for BWR HWC environments:

da/dt = $(2.72x10^{-8})*K^{2.5}$ in/hr for fluence from $5x10^{20}$ n/cm^2 to $3x10^{21}$ n/cm^2 at E>1.0 MeV

The NRC staff verified this CGR, including the applicable neutron fluence range, is the same as that recommended in BWRVIP-99-A, "BWR Vessel and Internals Project Crack Growth Rates in Irradiated Stainless Steels in BWR Internal Components" (Ref. 18).

Summary – Crack Growth Rate Models

EPRI's proposed CGR for both fluences have been approved for use in BWRs in the NRC staff's SEs of BWRVIP-14-A and BWRVIP-99-A. Both TRs consider HWC to be effective when the ECP is -230 millivolts versus the standard hydrogen electrode (mV_{SHE}), while typical PWR environments may have an ECP of -700 mV or lower. The NRC staff examined Figure 6-5 of MRP-227, Rev. 0, which provides limited IASCC CGR data for PWR environments, and concluded that most of the data points corresponding to a neutron exposure below roughly 5 displacements-per-atom (dpa) or $3x10^{21}$ n/cm^2 are bounded by the BWR HWC CGR curve. The NRC staff finds the use of the BWR HWC CGRs proposed for both fluence ranges to be conservative and acceptable because the HWC environment is electrochemically similar to the PWR reactor coolant environment, the BWR HWC CGRs bound the limited CGR data in a PWR environment, and the CGR for the fluence ranges have been approved for use in BWRs under HWC in the NRC staff's SEs of BWRVIP-99 and BWRVIP-14-A. EPRI agreed to incorporate the CGR for both neutron fluence ranges discussed above into the final version of the TR. The staff verified these CGR were incorporated where applicable in the revised evaluation procedures submitted by letters dated January 16, April 10, and July 8, 2014 (Refs. 5, 6, and 7).

The proposed revised evaluation procedures for core barrel welds specify the CGR from MRP-227-A, Equation 6-5 for welds with neutron fluence greater than or equal to $5x10^{20}$ n/cm^2. This CGR equation is the same as that specified in BWRVIP-99-A for the fluence range of $5x10^{20}$ n/cm^2 to $3x10^{21}$ n/cm^2 at E>1.0 MeV. Since the applicability of this equation is limited to $3x10^{21}$ n/cm^2 or less, but the fluence for some core barrel welds may exceed this fluence, in follow-up RAI 6 the staff asked EPRI what CGRs, or adjustments to the CGRs will be applied for areas with fluence greater than $3x10^{21}$ n/cm^2. In its May 5, 2015, response (Ref. 9), EPRI proposed to adjust the methodology to require the use and justification of a model applicable to these welds above $3x10^{21}$ n/cm^2 (E > 1.0 MeV). EPRI further stated that such models for stainless steel exposed to fluence greater than $3x10^{21}$ n/cm^2 (E > 1.0 MeV) are currently available from sources like EPRI proprietary technical reports, which are available for use by the individual licensee in developing plant-specific acceptance criteria. The staff found EPRI's response to RAI 6 acceptable since EPRI has proposed alternate CGRs for welds with higher fluences. However, since the NRC staff has not reviewed or approved CGR models for

fluences greater than $3x10^{21}$ n/cm^2, the staff will impose as a condition of the use of this methodology, that the CGR models used must be either previously approved by the NRC staff, or the basis for the model must be submitted along with the analysis (Condition 2). Since evaluations for B&W RVI components could potentially require crack growth calculations, Condition 2 is applicable to B&W design RVI as well as Westinghouse and CE design RVI.

With this condition, the NRC staff considers the CGRs proposed by EPRI for use in RVI flaw evaluations to be acceptable.

Fracture Toughness and Fracture Mechanics Methodology

In its response to RAI 29, Part 5, EPRI indicated that the fracture toughness values and evaluation methodologies to be used are consistent with those recommended in BWRVIP-76-A, "BWR Vessel and Internals Project, BWR Core Shroud Inspection and Flaw Evaluation Guidelines" (Ref. 19) and MRP-227, Rev. 0. In the responses to RAI 31, Part 1 and RAI 34, Part 2a, EPRI cited a 2009 conference paper by J. K. McKinley, et al. (Ref. 20), for the recommended fracture toughness values of 150 ksi√in for lower fluences and 34.6 ksi√in for fluences greater than 15 dpa. This value (34.6 ksi√in) is the saturation value at a neutron exposure of 100 dpa. The NRC staff reviewed the referenced paper, and agrees that the fracture toughness value for fluence greater than 15 dpa has been determined conservatively from the data. However, examination of Figure 4 of Reference 14 shows that the fracture toughness of 150 ksi √in recommended for the fluence range of $3x10^{20}$ n/cm^2 to $1x10^{21}$ n/cm^2 does not represent a lower bound to the data; therefore, in RAI B the NRC staff requested EPRI provide additional justification for the use of this fracture toughness value. Additionally in RAI B, the NRC staff asked EPRI to clarify which fluence range is considered "lower fluence." In its response to RAI B, EPRI indicated that the use of a fracture toughness value of 150 ksi√in will be limited to pre-inspection or generic evaluations for neutron fluences of 0.5 dpa ($3x10^{20}$ n/cm^2) or below, as an alternative to limit load analysis. For neutron fluences between $3x10^{20}$ n/cm^2 to $1x10^{21}$ n/cm^2, a fracture toughness of 112 ksi√in will be used for LEFM analyses as an alternative to elastic plastic fracture mechanics (EPFM). Additionally in response to RAI B, EPRI clarified that "low fluence" means fluence levels equal to or below 0.5 dpa ($3x10^{20}$ n/cm^2, E>1MeV). RAI B is therefore resolved.

In several RAI responses, EPRI referenced BWRVIP-76-A with respect to fracture mechanics evaluation methods and fracture toughness values. In RAI H the NRC staff requested clarification of which methodologies would be incorporated (MRP-227-A or BWRVIP-76-A), and in RAI C that the recommended methodologies and fracture toughness values be incorporated into the final version of the TR. In its responses to RAIs C and H, EPRI indicated that the fracture toughness methodologies recommended in MRP-227-A, Section 6 would be used. EPRI also provided tables summarizing the recommended methodologies. Two different tables with slightly different methodologies were provided for 1) pre-inspection and generic evaluations, and 2) for evaluation of flaws observed in-service.

The NRC staff compared the proposed methods and toughness values to be incorporated into the TR with the approved fracture toughness values and evaluation methods from BWRVIP-76-A. Appendix D to BWRVIP-76-A specifies evaluation methods as a function of fluence, and provides fracture toughness (K_{IC}) values as a function of fluence for LEFM.

The NRC staff notes that although the BWRVIP-76-A fracture toughness data were determined from materials irradiated under BWR conditions, the response of austenitic stainless steels to neutron irradiation in a PWR should be similar. Thus, based on the similarity in the neutron

spectra, the BWR data can be applied to a PWR until more PWR-specific fracture toughness data are accumulated.

The NRC staff reviewed the fracture mechanics methodologies and fracture toughness values proposed for incorporation into the TR and finds that they are conservative compared to the BWRVIP-76-A requirements. The conservatisms include use of a lower toughness value of 112 ksi√in over the entire fluence range of $3x10^{20}$ - $3x10^{21}$ n/cm^2 (E>1MeV), while the BWRVIP allows a higher toughness of 150 ksi√in up to $1x10^{21}$ n/cm^2. Also, the option to use LEFM at fluences less than or equal to 3 $x10^{20}$ n/cm^2 is conservative since austenitic stainless steel should retain ample plasticity in this fluence range. EPRI also proposed a lower toughness value for LEFM evaluations at fluences greater than 15 dpa ($1x10^{22}$ n/cm^2), which is consistent with the fracture toughness recommended in MRP-227-A. The staff verified the revised fracture mechanics methodologies as summarized in the RAI B, C, and H responses were incorporated where applicable in the revised evaluation procedures submitted by letters dated January 16, April 10, and July 8, 2014 (Refs. 5, 6, and 7). RAIs C and H are thus resolved.

Treatment of Flaws in Close Proximity

The NRC staff requested in several RAIs (29, 31, 34, 35, and 39) related to several different components, a discussion of how flaws in close proximity would be handled. EPRI's responses to these RAIs indicated that cracks in sufficiently close proximity such that it cannot be demonstrated that their growth rates are not affected by each other would have their lengths added and that the guidance for the determination of whether cracks are too close is consistent with the 2004 ASME Code, Section XI. The NRC staff finds this approach to be acceptable because it is consistent with the ASME Code, Section XI. In RAI D, the staff requested a description of the relevant ASME Code guidance and a reference to the ASME Code guidance be added to the methodology of the items to which it applies. In its January 3, 2013, response to RAI D, EPRI indicated that the basis for the guidance for flaws in close proximity is ASME Code Section XI: IWA-3330(a) and Figure IWA-3330-1, which provide requirements for the minimum allowable separation distance, and that a description of and reference to this guidance will be included in the relevant sections in the updated TR. RAI D is thus resolved. The staff verified the guidance on flaw proximity as summarized in the RAI D response is incorporated where applicable in the revised evaluation procedures submitted by letters dated January 16, April 10, and July 8, 2014 (Ref. 5, 6, and 7).

Components Qualified via a Time-Limited Aging Analysis

MRP-227, Rev. 0, required inspections of several components for fatigue cracking only if the fatigue life of the component could not be qualified via a TLAA.

The items[1] listed below are susceptible to cracking due to fatigue and are only inspected if they cannot be qualified via a TLAA:

- CE-ID: 9 Core Support Barrel Assembly –Lower Flange Flexure Weld
- CE-ID: 10 Lower Support Structure – Core Support Plate
- CE-ID: 11 Upper Internals Assembly – Fuel Alignment Plate
- CE-ID: 13 Lower Support Structure – Deep Beams

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[1] The staff notes that the ID numbers used in the TR and some of the component names were changed in EPRI's letters dated April 10 and July 8, 2014. The changes were made to align with MRP-227-A. The ID numbers and component names in this SE reflect the updated designations.

If fatigue life cannot be demonstrated via a TLAA, the components are subjected to an EVT-1 examination. The examination scope is determined by the results of the plant-specific fatigue analysis.

For CE-ID: 9, the analysis goal is to demonstrate that observed flaws are not actively growing. For the other three components, the analysis goals focus on demonstrating continued functionality of the components. For all four components, fatigue CGRs based on the stress amplitudes from the TLAA are used to establish acceptance criteria for the inspections. The approach for all four components is plant-specific.

Since CE-ID: 10 and CE-ID: 11 have similar functions, in RAI 36, the NRC staff requested a discussion of whether a similar analysis procedure should be used for both components, since a much more detailed procedure was included in the TR for CE-ID: 8. In its response, EPRI provided some process steps for an analysis procedure for CE-ID: 9. The response also stated that Appendix C Item CE-ID: 9 will be revised to incorporate the process used to determine allowable crack length described in the RAI response. The staff verified that the additional detail is included in the revised methodology for the fuel alignment plate provided in EPRI's January 16, 2014, letter (Ref. 5).

Under "Acceptance Criteria" the TR states, "Acceptance criteria for TLAA related items are beyond the scope of the current project." However, if the items are being inspected, it is because the fatigue life could not be qualified via a TLAA for the life of the plant. Therefore, in RAI 41, the NRC staff requested that EPRI discuss whether it is possible to define acceptance criteria based on ensuring structural integrity of the components until the next inspection. In its response to RAI 41, EPRI stated that a general approach for determining these acceptance criteria is outlined for each of the components under the "Analysis" section. Such an acceptance criteria would be calculated on a plant-specific basis and would account for code requirements, expected crack shapes, expected fatigue usage, and operating loads. The basis used to determine that the projected crack growth is acceptable will be added to the Acceptance Criteria section for each component. EPRI further stated under "Actions" that the acceptance criteria for the affected components would be revised in conjunction with the revisions made to address RAI 36. The staff verified that the revised methodology provided for the components in EPRI's letters dated January 16 and April 10, 2014 (Refs. 5 and 6) includes acceptance criteria for the affected components. For the Lower Support Structure – Core Support Plate and the Upper Internals Assembly – Fuel Alignment Plate, functionality based acceptance criteria are provided. The NRC staff finds EPRI's responses to RAI 36 and RAI 41 acceptable because the proposed methodology and acceptance criteria will ensure the component function is preserved if qualification of the fatigue life via a TLAA is unsuccessful.

In RAI 37, the NRC staff requested EPRI address the case of multiple cracked beams for CE-ID: 11, and justify allowing one whole beam in the deep beam structure to be completely cracked. Summarizing EPRI's response, the methodology assumed only one crack will occur in one beam, because the probability of even one cracked beam was considered sufficiently unlikely that the assumption of multiple failures is unnecessary, and one completely cracked beam is acceptable because of the high degree of redundancy in the lower support structure. The NRC staff finds EPRI's response acceptable, thus RAI 37 is resolved.

The proposed revision to the evaluation methodology for the core support barrel assembly – lower core barrel flange welds and lower support structure - deep beams provided in EPRI's April 10, 2014, letter contain extensive changes compared to the original methodology. The revised methodology includes specification of the BWRVIP CGR and ASME Code proximity rules in the data requirements for a fracture mechanics evaluation (if EVT-1 inspection is performed instead of a TLAA). In the case that a TLAA is performed, the revised methodology also provides additional guidance for performing the fatigue analysis, including the incorporation of environmental effects in accordance with NUREG/CR-6909.

The revised evaluation procedures for CE-ID: 10 Lower Support Structure – Core Support Plate and CE-ID: 11 Upper Internals Assembly – Fuel Alignment Plate, provided in EPRI's January 16, 2014, letter, include acceptance criteria for evaluation of cracks, but do not contain the additional details for the TLAA evaluation that were included for the core support barrel assembly – lower core barrel flange welds and lower support structure - deep beams. It was not clear to the staff that guidance for performing TLAAs in lieu of inspection for certain components is within the scope of Section 1 of the TR. Therefore, in follow-up RAI 3 dated February 11, 2015, the staff requested that EPRI: (1) clarify whether guidance for such TLAAs is within scope of the TR; (2) modify the "Objective" section of the TR accordingly; (3) if such guidance is within the TR scope, modify the guidance to the evaluation procedures to include the guidance for the TLAA evaluation; and (4) if guidance for TLAA's is not within the scope of WCAP-17096-NP, remove this guidance from the evaluation procedures for the two components. In its May 5, 2015, response to follow-up RAI 3 (Ref. 9), the MRP indicated it is not the intent of WCAP-17096-NP to provide guidance on evaluating TLAAs, and provided markups of the evaluation procedures for the two components removing this guidance. The staff reviewed the markups for the two components, and found the changes to be acceptable. The issue in follow-up RAI 3 is therefore resolved. The procedures for evaluation of cracks for these components are discussed in Section 3.2.2.2.2 under the heading "Other Core Support Structures."

3.2.2.2.2 Component-Specific Methodology and Data Requirements

This SE section describes the NRC staff's evaluations of concerns specific to particular component types. Since Appendix C and Appendix E of the TR contain separate methodologies for numerous Westinghouse and CE Primary and Expansion component items, to simplify its evaluation, the NRC staff grouped these components based on similar form, function, and evaluation methodology. In the following sections, CE RVI components have component designators CE-ID: X and Westinghouse RVI components have component designators W-ID: X, as used in the TR.

<u>Bolting Components</u>

This category includes the following:

- CE-ID: 1, Core Shroud Assembly (bolted) - Core shroud bolts
- CE-ID: 1.1, Core Shroud Assembly (bolted) - Barrel-shroud bolts
- W-ID: 7, Baffle-former assembly - Baffle former bolts
- W-ID: 7.1, Core Barrel Assembly - Barrel-former bolts

The MRP-227-A report recommends UT examination for these RVI components subjected to the aging effect of cracking caused by IASCC and/or fatigue. Data requirements defined by the TR for these items include bolting patterns, shroud design, fast neutron (dpa) distribution in the core shroud, projected bolt failure rate, and minimum bolting pattern analysis.

The TR indicated that a generic approach would be used for W-ID: 5 and W-ID: 5.1. For CE-ID: 1 and CE-ID: 1.1, the TR does not recommend a generic approach since only two CE plants have these bolts.

The TR requires that the inspection results have an acceptable bolt pattern and margin to protect against additional failures. The margin is defined as:

$M = N - N_{req} – N_f$, where

N = total number of baffle-former bolts
N_{req} = number of baffle-former bolts in the acceptable pattern with only required number of bolts
N_f = number of failed bolts

For simplicity, the acceptable pattern with only the minimum required number of bolts will be referred to as the acceptable bolt pattern throughout the SE. Establishment of N_{req} is discussed later. The TR states that at the time of the first inspection, for operation through the next 10 – 15 effective full power year (EFPY) interval, it is acceptable if no more than 50 percent of initial margin is consumed. Although the TR states that the acceptable bolt pattern is outside the scope of the TR, the acceptance criteria for several of the bolting components refer to WCAP-15030-NP-A, “Westinghouse Methodology for Evaluating the Acceptability of Baffle-Former-Barrel Bolting Distributions under Faulted Load Conditions” (Ref. 21), for the acceptable bolt pattern of observed intact bolts. Therefore, in RAI 32, Part 1, the NRC staff requested EPRI describe how N_{req} was determined including whether a generic methodology previously submitted to the NRC was used.

The NRC staff issued RAI 32, Part 2, requesting EPRI to justify the margin criterion for failed bolts. In the acceptable bolt pattern, since a simple numerical margin may not be appropriate, in RAI 32, Part 3, the staff requested EPRI discuss whether the acceptable bolt pattern or distribution needs be considered when determining the adequacy of the remaining margin. Regarding MRP-227-A’s recommendations for examination coverage of baffle-former bolts, RAI 32, Part 4, further requested EPRI explain: (1) how N_f is determined, and (2) what assumptions are made regarding the inaccessible bolts. RAI 32, Part 5, requested information on potential undetected flaws in bolts due to limitations in the inspection technology. RAI 33 Part 2 addressed similar issues, thus the responses are evaluated together under the evaluation of RAI 33, Part 2, below. RAI 32 also asked whether failed bolts must be replaced (Part 6), the potential need for destructive examination and replacement of failed bolts (Part 7), a summary of OE related to failures of the various internals bolting (Part 8), and requested that the condition from the MRP-227 final SE regarding the maximum re-inspection interval of 10 years be addressed (Part 9).

In response to RAI 32, Part 1, EPRI indicated that N_{req} is determined via an Acceptable Bolting Pattern Analysis, using the NRC-approved methodology outlined in WCAP-15029-P-A, Rev. 1, "Westinghouse Methodology for Evaluating the Acceptability of Baffle-Former-Barrel Bolting Distributions under Faulted Load Conditions" (Ref. 22). EPRI also stated that the affected component evaluations will be revised to reference WCAP-15029-P-A, Rev. 1. The staff verified that the proposed revision to the bolting methodology in EPRI's January 16, 2014, letter (Ref. 5) include this reference. The NRC staff finds the response to RAI 32, Part 1, acceptable because EPRI will reference an NRC-approved methodology for determining the required number of intact baffle-former bolts.

In response to RAI 32, Part 2, EPRI indicated that the basis for the acceptance criteria that no more than 50 percent of the margin be consumed is an assumption that the number of failures in the next 10 years will be no more than the number of failures observed to date. EPRI also indicated that it is not possible, based on the limited number of failures of the RVI bolting to date, to create a probabilistic model for prediction of bolt failure rates, and that no specific distribution of failures as a function of EFPY was assumed in the development of the margin term. However, EPRI also noted that MRP-227 calls for the first inspection of the baffle-former bolts between 25 and 35 EFPY of plant operation, and that the TR margin requirement that no more that 50 percent of the bolts outside the acceptable bolt pattern have failed at the time of the initial inspection essentially allows the maximum number of bolts that failed during the 10-year inspection interval to be equal to the number of bolts that failed in the first 40 years of operation. EPRI further noted that while this projection methodology allows for a slight acceleration of the bolt failure rate, the conditions for swelling and bolt failure may be less severe later in plant life due to conversion to low-leakage core loading patterns.

The NRC staff finds EPRI's response to RAI 32, Part 2, acceptable because the assumption that the same number of bolts fail in the 10 calendar years subsequent to the inspection as in the previous (25 EFPY minimum) implies an increase in the average failure rate by a factor of at least 2.5. The NRC staff finds this is conservative, since the conversion to low-leakage core loading patterns should result in a decrease in the IASCC failure rate later in plant life.

In response to RAI 32, Part 3, EPRI indicated that the acceptance criteria of the TR require that both the required number of bolts and an acceptable bolt pattern be met, thus ensuring that there would not be a significant number of failures on one baffle plate. The NRC staff finds this response acceptable because EPRI's acceptance criteria require both a margin and the acceptable bolt pattern be met, thus, maintaining the structural stability of the baffle assembly.

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 uninspected population would be added to the number of failed bolts in the inspected population to determine the overall failed population N_f. The NRC staff finds this response acceptable because it provides a conservative means of estimating the total number of failed bolts.

In response to RAI 32, Part 6, EPRI indicated that no replacement of failed bolts is required if the remaining intact bolts satisfy the acceptance criteria of the established acceptable bolt pattern and sufficient margin of intact bolts exists. The NRC staff finds this response acceptable because the required margins of intact bolts and structural stability of the RVI will be maintained.

The EPRI's response to RAI 32, Part 9, stated that the TR will be modified as necessary to maintain consistency with MRP-227-A. The NRC staff finds this acceptable because it will ensure the MRP-227 SE condition regarding a 10-year maximum re-inspection interval is addressed in the final version of the TR. The staff verified this change was incorporated in the revised evaluation procedures provided in EPRI's January 16, 2014, letter (Ref. 5).

RVI bolting is susceptible to loss of fracture toughness due to irradiation embrittlement. Due to the detectability limitations on the UT examination technique for the RVI bolting, the NRC staff issued RAI 33 requesting EPRI discuss (1) how irradiation embrittlement is accounted for in the determination of the acceptable bolt pattern and (2) how potential undetected flaws in bolts are accounted for in the determination of the acceptable bolt pattern.

EPRI's response to RAI 33, Part 1, indicated testing of irradiated bolts have shown sufficient ductility such that the ASME allowable stresses can still be used. EPRI cited testing of bolts removed from Farley that demonstrated good ductility. EPRI indicated that the maximum fluence value of the Farley baffle-former bolts was approximately 10 dpa ($7x10^{21}$ n/cm^2, E > 1MeV), approximately 20 percent of the anticipated end-of-life fluence of the bolts, but changes in the mechanical and fracture properties occur most rapidly between 1 and 5 dpa and saturate by 10 dpa. Therefore, the Farley bolt tensile results can be considered representative with respect to end-of-life properties. EPRI cited test data presented in MRP-175 (Ref. 23) and "Materials Reliability Program: PWR Internals Age-Related Material Properties, Degradation Mechanisms, Models, and Basis Data - State of Knowledge," dated December 31, 2007 (MRP-211) (proprietary report – non-publically available) to support this response. The NRC staff reviewed the referenced test data and agrees with EPRI's conclusion. RAI 33, Part 1, is thus resolved.

EPRI's responses to RAI 33, Part 2, and RAI 32, Part 5, are very similar and can be summarized as follows: Undetected flaws in bolts would likely grow quickly and result in bolt failure within one to two cycles. Therefore, a fracture mechanics approach to demonstrate undetected flaws would be stable until the next inspection is not viable. EPRI stated that laboratory data on IASCC clearly indicates that crack initiation is the controlling step in this process, which implies that observed bolt failure rates provide a reasonable indication of the crack initiation rate. The potential for undetected flaws is therefore accounted for in the margin of intact bolts, which is designed to account for both initiation of new flaws and growth of existing undetected flaws. The NRC staff finds this response acceptable because the NRC staff found the acceptance criteria of no more than 50 percent of the margin consumed is conservative, as discussed in the NRC staff's evaluation of RAI 32, Part 2.

EPRI's responses to RAI 32, Part 8, and RAI 33, Part 3, provided a summary of OE related to previous bolting inspections. The EPRI's summary is consistent with the information on RVI bolting OE in MRP-227-A, Appendix A. Bolting inspection OE suggests that the rate of defective bolts at the first inspection is less than 10 percent, and is less than 1 percent for inspections conducted since MRP-227, Rev. 0 was issued. EPRI's responses note that the acceptance criterion of no more that 50 percent of the margin of required bolts consumed and of a linear failure rate takes into account the small numbers of failed bolts detected in previous inspections.

The NRC staff finds the proposed approach for determining the acceptance criteria, specifically the acceptable bolt pattern, acceptable because the approach will ensure appropriate structural margins exist for all AOOs and design basis events, with respect to the number of intact bolts, and justifies the required margin of intact bolts based on OE. In addition, the acceptable bolt

pattern accounts for reduced fracture toughness of the bolts due to irradiation, and NDE detectability limitations.

W-ID: 4(6) – Baffle-Former Assembly – Baffle-edge Bolts

Unlike the other bolts discussed above, MRP-227-A does not require UT examinations of the baffle-edge bolts but specifies a VT-3 examination for these bolts. The relevant degradation mechanism is cracking due to IASCC and fatigue. The TR indicates that studies have shown these bolts are not required for structural integrity of the baffle. However, the edge bolts prevent gaps between plates that can lead to baffle jetting that can result in damage to peripheral fuel assemblies. The acceptance criteria for these bolts are therefore to be determined via a failure modes and effects analysis (FMEA).

In RAI 42, the NRC staff requested EPRI discuss whether the baffle-edge bolt FMEA would be part of the overall baffle-former assembly FMEA, whether gaps between plates would be considered, how undetected failed bolts would be accounted for since a UT examination will not be performed on these bolts, and details on the FMEA process for these bolts. In its response to RAI 42, EPRI answered “yes” for the first two concerns. In response to RAI 42, Part 3, concerning the possibility that failed baffle-edge bolts may not be detected if the locking bars remain intact; EPRI indicated that loss of integrity of baffle-edge bolts without a corresponding loss of integrity of the baffle-former bolts would not result in a loss of function of the baffle-former assembly, including prevention of baffle jetting. Further, EPRI stated the primary concern (related to baffle-edge bolt failure) is the generation of loose parts. Therefore, the VT-3 inspection will provide reasonable assurance that any baffle-edge bolt will not become a loose part. In response to RAI 42, Part 4, EPRI indicated that the FMEA process would be essentially the same as that described in MRP-191 (Ref. 12). Also in response to Part 4, EPRI provided detail on the process steps to a FMEA, relevant conditions that may be considered for the baffle-former assembly FMEA, and possible consequences, to be added to the TR. The NRC staff finds EPRI’s response to RAI 42, Parts 1 and 2, acceptable because EPRI has clarified that the baffle-edge bolt VT-3 examination results will be evaluated in the overall context of the baffle-former assembly and the observation of gaps that can allow baffle-jetting will be considered in the FMEA. The NRC staff finds EPRI’s response to RAI 42, Part 3, acceptable because the FMEA baffle-edge bolts are not essential to preserving the function of the baffle-former assembly, including prevention of baffle-jetting, and VT-3 examination is sufficient to detect the general failure of the baffle-edge bolts. The NRC staff finds EPRI’s response to RAI 42, Part 4, acceptable because it clarifies how the FMEA will be conducted for the baffle-former assembly, including the baffle-edge bolts. The staff verified that the guidance for conducting the FMEA has been incorporated into the revised evaluation procedures provided for the baffle-edge bolts (renumbered W-ID: 6) in EPRI’s January 16, 2014, letter, as EPRI committed in response to RAI 42, Part 4. RAI 42 is therefore resolved.

The NRC staff finds the use of an FMEA to define the acceptance criteria for the baffle-edge bolts to be acceptable because this will ensure that the function of the bolts will be preserved, and the function of the baffle-edge bolts will be considered as part of the overall baffle-former assembly FMEA. The TR, under “Approach,” states that the FMEA should address plant-specific practices and priorities, and that some generic work is possible to outline issues and options to be addressed in the FMEA. The NRC staff will impose a condition (Condition 3, Group 1) for plant-specific application of this TR that any generic or plant-specific FMEA methodology be submitted to the NRC to determine whether review and approval is needed,

within one year[1] after any inspection that detects relevant conditions for the baffle-edge bolts as defined in the "Examination Acceptance Criteria" column of Table 5-3 of MRP-227-A.

Lower Support Structures – Columns, Welds, and Bolts

The TR proposes a similar methodology for the following components:

- CE-ID: 1.2 Core Shroud Assembly (Bolted) - Core Support Column Bolts
- CE-ID: 8 Lower Support Structure – Core Support Column Welds
- W-ID: 2.3 – Lower Support Assembly – Lower Support Column Bodies (Cast),
- W-ID: 3.2 – Lower Support Assembly – Lower Support Columns (Non- Cast)
- W-ID: 7.2 – Lower Support Assembly – Lower Support Column Bolts

These support structures are generally susceptible to cracking due to IASCC plus SCC and fatigue for some components. Similar to bolting, it is possible to define an acceptable pattern of intact columns that is necessary to perform the structural function of supporting the core plate. The analysis of the acceptable column pattern generally involves a finite element analysis (FEA) of the core plate and support columns to determine the acceptable pattern(s). The components are inspected using UT (for bolts), EVT-1, or VT-3 examination. Columns or bolts found to be cracked are generally assumed to be completely failed for analysis purposes.

The data requirements include, as a minimum, the loads on the lower core plate and the displacement tolerances on the lower core plate. Most of the components also take into account the irradiated material properties of the stainless steel. For the Westinghouse cast core support columns, thermal aging embrittlement is also taken into account. In RAI 38, the NRC staff requested that EPRI discuss whether the CE core support column bolts (CE-ID: 1.2) and Westinghouse lower core support column bolts (W-ID: 7.2) data requirements should take into account the irradiated material properties. In its response, EPRI stated that "Data Requirements" section of Appendix C Item CE-ID: 1.2 and Appendix E Item W-ID: 7.2 would be updated to include the fluence accumulated by the core support column bolts/lower support column bolts and a model for stainless steel properties as a function of irradiation. The staff verified this change was incorporated in the revised evaluation procedures for CE-ID: 1.2 and W-ID: 7.2 provided in EPRI's January 16, 2014, letter (Ref. 5). The NRC staff finds the response to RAI 38 acceptable because it will ensure that irradiated material properties are accounted for in the evaluation of the core support column bolts.

The analysis procedures for the items listed above provide the following method of determining the margin:

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 acceptable column pattern

Margin = $N - N_{req}$

The following acceptance criterion is given:

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[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.

Require that no more of 1/2 of columns in margin are failed:

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

In RAI 27, Part 1, the NRC staff requested EPRI provide additional information regarding the method of determining the acceptance criterion for failed column bodies. Additionally, since the location as well as the number of intact columns and bolts may affect the acceptance criterion, a simple numerical margin may not be appropriate. In RAI 27, Part 2, the NRC staff therefore requested that EPRI discuss whether the acceptance criterion needs to consider the location of the additional intact columns and bolts that constitute the margin.

EPRI's response to RAI 27, Part 1, states that the basis for the given acceptance criterion is an assumption that the failures in the next 10 years will be equal to the number observed during the first inspection, at approximately 40 calendar years of operation. No specific distribution was assumed in the development of the margin because there have been no failures to date on which to base any kind of statistical model. The NRC staff finds EPRI's proposed acceptance criterion is reasonable because it allows for an acceleration of the failure rate by a factor approaching four (dependent on the EFPY at the time of the inspection and the plant capacity factor over the subsequent 10 years).

In response to RAI 27, Part 2, EPRI indicated that once the acceptable column pattern is established, the remaining intact columns are by definition, outside the acceptable column pattern, and that it is not practical for the acceptance criteria to be based on a predicted failure pattern. The response to RAI 27, Part 2, further stated that the intact columns and bolts should be well-distributed among the core support columns across the core plate, the acceptance criteria already require a minimum distance between failed columns, and the acceptance criteria are conservative because any cracked column is considered failed. The NRC staff finds the response to RAI 27, Part 2, acceptable because a minimum distance between cracked columns should ensure that columns or bolts constituting the margin are relatively evenly distributed. The approach for the support columns, and associated bolts or welds would involve a pilot analysis of lower support structure to identify critical issues. The TR states that final acceptance would be expected based on plant-specific analysis. The NRC staff therefore imposes a condition (Condition 3, Group 2) requiring a licensee using this methodology submit their plant-specific analysis for the acceptable minimum distribution of intact core support columns, and/or their associated bolts and welds, for 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 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:

[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.

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

Applicant/Licensee Action Item 7 from the December 16, 2011, SE in MRP-227-A, regarding plant-specific functionality analyses for CASS components, required licensees to develop plant-specific analyses demonstrating that the CE lower support columns and Westinghouse lower support column bodies (cast) will maintain their functionality during the period of extended operation. If these plant-specific functionality analyses have been performed and submitted to the NRC staff prior to the initial inspection of the components, licensees may be able to compare the inspection findings to the functionality analyses and may not need to develop this plant-specific analyses unless the inspection findings are not bounded by the functionality analysis.

The NRC staff found the analysis approach for the support columns, and associated bolts and welds to be acceptable because the approach for defining the acceptable column pattern uses appropriate engineering techniques, takes into account the appropriate inputs, and will be reviewed on a plant-specific basis for different plant designs.

<u>Core Barrel and Core Support Barrel Flanges and Welds, CE Lower Core Support Beams</u>

Both the CE and Westinghouse RVI designs have core barrels that are welded structures. CE-ID: 6.4 Lower Support Structure – Lower Core Support Beams are also discussed under this heading because similar evaluation procedures are used. The component identifications are:

- CE-ID: 6 Core Support Barrel Assembly – Upper (Core Support Barrel) Flange Weld
- 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 Core Support Barrel Assembly – Lower Cylinder Girth 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: 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
- 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

The NRC staff grouped these items together for evaluation because they are mostly welds or major structural components associated with the core barrel, and all use similar fracture mechanics-based evaluation techniques. MRP-227-A generally requires an EVT-1 examination of 100 percent of the accessible surfaces of these welds. For the welds in Westinghouse-design RVI, MRP-227-A specifies examination of one side of the weld only. For these welded structures, the TR generally proposes a fracture mechanics-based approach for evaluating flaws. The relevant aging effect for these welds is generally SCC and possibly fatigue, except for welds in the beltline region, which can also crack due to IASCC. The stated goal of the evaluation for all these welds is to demonstrate the crack is stable or not likely to grow through wall. The TR states under "Goal," that "Due to the high fracture toughness of unirradiated stainless steel, the core barrel is a highly flaw tolerant structure and [the allowable] flaw sizes

are expected to be very large." The data requirements for these components generally include operating loads, applied stress intensity factor (K_{app}) solutions for range of expected crack shapes, estimated neutron fluence, and SCC and fatigue CGR curves. The NRC staff finds the data requirements are appropriate because they provide the necessary data to evaluate SCC, and fatigue.

The analysis methodology described for these welds in the TR relies on LEFM. Two different fracture toughness values are used depending upon the specific location in the core support barrel. The NRC staff issued several RAI's to clarify inconsistencies in the methodology for the different welds, and the basis for fracture toughness values used (RAIs 29, 31, 35, 39, 43, and 49). With respect to the consistency of analysis procedures, EPRI stated that the analysis procedures would be modified to be consistent with BWRVIP-76-A (Ref. 19), as evidenced by the proposed edits for all the core barrel weld components listed above provided in EPRI's letters dated April 10 and July 8, 2014 (Refs. 6 and 7). The basis for the fracture toughness values used for evaluations of detected flaws is also addressed in BWRVIP-76-A.

As noted above, MRP-227-A allows visual examinations to be performed from one side of the weld, and does not require supplemental UT examination to determine flaw depth. Under "Inputs and Assumptions," the revised procedures include the following:

> "The inspections identified in MRP-227-A 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 stress corrosion cracking (SCC)."

Under "Data Requirements," the procedures, with respect to "Flaw Depth," state (in part) that:

(a) For one-sided visual inspections, the flaw is assumed to be through-wall, and
(b) Supplemental examinations may be used to determine flaw depth for a flaw-specific criterion, if needed.

Also under "Data Requirement," Item 4 states (in part) (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, and (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 ultrasonic testing examinations.

Under analysis, the procedures state, in part, that:

> "All analyses require an assumption of the SCC/ IASCC crack growth expected over the upcoming period of service. The methodology is based on analysis of a through-wall flaw with weld residual and thermal stresses relieved."

Since supplementary inspections to determine the crack depth are allowed but not required by the evaluation procedure, the staff notes that a visually observed crack may not be through-wall. The staff was also concerned that, considering that weld residual and thermal stresses would still act on a non-through-wall crack, the assumption that the CGR for a through-wall crack would bound that of a non-through-wall crack may not be conservative. Therefore, in follow-up RAI 1 dated August 18, 2014 (Ref. 25), the staff asked EPRI to demonstrate that a non-through-wall crack will not grow at higher rate such that it would attain critical size prior to the next scheduled inspection. Follow-up RAI 1 also stated that this demonstration may be done generically or the evaluation procedures may be modified to require such a demonstration as part of the evaluation

of the specific weld in which degradation has been detected.

In its October 31, 2014, response to follow-up RAI 1, EPRI provided a qualitative discussion of why it believes the through-wall assumption is adequate. The relevant points of EPRI's response are summarized as follows:

- Mechanical failure of the core barrel girth welds is based on the limiting solution provided by a through-wall flaw with a length equal to the observed surface length. For equivalent external loading, the type of crack with the maximum applied stress intensity factor for circumferential cracking is the through-wall crack.
- For typical weld residual stress distributions, which are compressive in the mid-wall, the conditions favor accelerated surface CGRs and the formation of high aspect ratio flaws. Through-wall crack growth is not anticipated in flaws when the surface length is comparable to the barrel thickness.
- Evaluation of crack growth in a non-through-wall crack would require both supplemental inspections to determine crack depth and complicated FEA models to simulate the relaxation of the secondary thermal and weld residual stresses as the crack grows.
- As an alternative to the supplemental inspection and FEA model, a crack growth verification requirement was included in the WCAP-17096 core barrel weld acceptance criteria methodologies. The intent of this verification requirement is to assure that the through-wall flaw analysis will remain conservative through the allowed inspection interval. Observation of CGRs exceeding the predicted rate would require further evaluation of the specific weld in which the degradation has been detected. The observation of unexpectedly high CGRs would trigger a reassessment of these criteria.

The staff found that EPRI's response to follow-up RAI 1 was not adequate to address the staff's concern that residual and thermal stresses acting on a non-through-wall crack could lead to a higher CGR resulting a crack that could challenge structural margins sooner than if the detected cracks is assumed to be through-wall, but with a lower CGR due to the absence of residual and thermal stresses.

Under "Analysis," the procedures state that in order to apply the acceptance criteria to a full 10-year inspection interval, follow-up action is required to verify the assumptions used in the predicted CGR, and that 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. However, the staff was concerned this verification guidance was not sufficiently prescriptive, because it did not specify or justify the time frame for the verification inspection. Therefore, in follow-up RAI 2, the staff requested EPRI to modify the evaluation procedures to address the following:

a. Define the re-inspection time (e.g. the first refueling outage (RFO) after the initial inspection) and describe how the required re-inspection time is determined.

b. Provide details about any follow-up action other than re-inspection, and justify how such actions could verify the predicted CGR.

In its October 31, 2014, response to follow-up RAI 2 (Ref. 8), Part a, EPRI stated that the evaluation procedures will be modified to state a default verification period of one fuel cycle and that a technical justification will be required for any verification period greater than one fuel cycle. EPRI further stated that one fuel cycle is the shortest practical re-inspection time for the verification process, and that the acceptance criteria methodology in WCAP-17096-NP requires an allowance for 10 years of crack growth in a through-wall flaw. EPRI also stated that completion of the verification within 1-2 years is expected to provide sufficient margin for higher

than anticipated crack growth but must be logically assessed against what is actually detected during the examination so that core barrel removals are minimized.

EPRI further proposed in the response to follow-up RAI 2, to add the following wording to the "Analysis" section for the affected components:

> "In order to apply the acceptance criteria to a full 10-year 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."

In response to Part b of follow-up RAI 2, EPRI stated that "initial applications of this methodology have required a follow-up re-inspection to verify the crack growth rate." The response further stated that "It is recognized that follow-up actions other than re-inspection may involve supplemental examinations, testing or more complex analysis using updated crack growth models," and that "such alternatives would require technical justification."

The staff was concerned that even a one-cycle re-inspection interval may not be sufficiently conservative given that the inspection of the core barrel welds is only required to be performed visually from one side of the weld. Therefore, an observed crack may not be through-wall, thus the CGR could be affected by weld residual stresses and thermal stresses.

In follow-up RAI 5, the staff requested EPRI to explain what CGR model is used to determine the timing of the re-inspection interval. The staff also requested in RAI 5 that EPRI justify that the assumed CGRs are sufficiently conservative to account for uncertainties in CGR, which may result from the action of residual or thermal stresses on a non-through-wall crack.

In its May 5, 2015, response to RAI 5 (Ref. 9), EPRI indicated that the K-dependent BWR HWC CGRs will be used to determine the timing of the re-inspection interval as well as the allowable crack size for 10 years of operation. These are the same CGRs discussed in Section 3.2.2.2.1 of this SE. EPRI stated the CGR is conservatively based on the fluence at the end of the 10-year inspection interval. However, the BWR HWC CGRs are only valid up to neutron fluence less than or equal to $3x10^{21}$ n/cm^2. Therefore, EPRI indicated that an alternative CGR model would be used for fluences exceeding $3x10^{21}$ n/cm^2.

In response to the second part of RAI 5, EPRI provided arguments to justify the conservatism of the CGR models to be used. EPRI stated that the proposed approach evaluates the flaw as a through-wall flaw, and that this means that for a given set of loading conditions, mechanical stresses will generate a smaller critical flaw size than would be calculated using a part-through semi-elliptical flaw solution. EPRI indicated that secondary stresses, such as weld residual and thermal stresses, would have the greatest effect on shallow cracks and would be relieved as the cracks grow through-wall. EPRI also indicated that although these secondary stresses could increase the CGR of shallow cracks, the critical flaw sizes for shallow non-through-wall cracks would be larger than those for through-wall cracks.

EPRI also noted in its response that if the follow-up action for crack growth verification was a follow-up visual examination at the next RFO after discovery of the crack, the 18-month interval represents only about 15 percent of the 10 year period. EPRI claims this is conservative since the allowable crack size at detection accounts for 10 years of crack growth, providing considerable margin.

EPRI also pointed out several other sources of additional margin beyond the crack growth model that help assure that the observed crack will not grow beyond a critical flaw size during one fuel cycle interval, including:

- The conservative critical flaw size generated using the through-wall flaw assumption.
- The methodology employs conservative fracture toughness values.
- The methodology contains an allowance for 10 years of growth in a through-wall flaw.
- The methodology does not credit time required for the flaw to grow through-wall.
- The maximum allowable crack size predicted by LEFM (the suggested method) provides conservative results compared to the other available methods such as EPFM or limit load.

The staff reviewed EPRI's responses to RAI 5, and finds EPRI has provided reasonable arguments supporting its assertion that a follow-up re-inspection at the next RFO (18 months) after the discovery of a crack, is conservative. In particular, the allowance of 10 years of crack growth in the allowable flaw sizes, while re-inspecting after just 15 percent of this time period, allows for an actual CGR to be 6.7 times higher than assumed. For plants on a 24-month refueling cycle, the reinspection would be at 20 percent of the 10-year period, allowing an actual CGR 5 times higher than assumed, which is still very conservative. In addition, the staff agrees that the fracture toughness values and use of LEFM methodology is conservative. Not crediting the time to grow through-wall adds conservatism if the flaws are non-through-wall (which are the flaws the staff was concerned about).

As discussed in Section 3.2.2.2.1, in response to follow-up RAI 6, EPRI stated it would use alternative CGR models for components with neutron fluences greater than $3x10^{21}$ n/cm^{2}. Condition 2 of this SE addresses the need for submittal of the basis for these alternative models if these models have not been previously approved by the staff.

However, since the proposed change for the affected components is not definitive regarding the need to perform a follow-up inspection no later than one RFO after discovery of the indication, the staff will require as a condition on the use of this methodology, that if only visual examination is performed, that a follow-up re-inspection will be performed no later than one RFO after discovery of the crack, unless the licensee provides a technical justification for a longer re-inspection interval for review and approval by the staff (Condition 3, Group 3, Item a). With the addition of this condition, the staff's concerns in follow-up RAI's 1, 2, and 5 are resolved.

The CE Core Support Barrel Assembly - Upper (Core Support Barrel) Flange Weld and Lower Support Structure - Lower Core Support Beams were moved from the No Additional Measures category to the Expansion category by Condition 1 of the staff's final SE of MRP-227, Rev. 0. Evaluation procedures for these components were provided in EPRI's April 10, 2014, letter (Ref. 6), as committed in EPRI's response to RAI 45. These evaluation procedures are similar to the revised procedures for the core barrel welds and flanges.

The NRC staff finds the proposed analysis procedures for the core barrel welds, flanges, and the CE lower core support beams to be acceptable because they are based on an NRC-accepted methodology for evaluation of BWR core shrouds, which are similar structures constructed from similar materials to the PWR core barrels, flanges, and support beams. The fracture toughness properties as a function of neutron fluence, and the recommended analysis

methods for different fluence ranges, should also be applicable to PWR core barrels, flanges, and support beams. Since the approach calls for plant-specific analysis, the NRC staff will impose as a condition (Condition 3, Group 3. Item b) that plant-specific crack growth and fracture mechanics analyses justifying the detected flaws be submitted 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 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

<u>CE Core Shroud Welds and Assembly</u>

Most CE-design plants have a core shroud that was assembled by welding, in contrast to the Westinghouse-design plants that have a bolted core shroud assembly. The component identifications are:

- 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 - Full Height) - Shroud Plates
- CE-ID: 3.1 Core Shroud Assembly (Welded) – Remaining Axial Welds, Ribs and Rings

The NRC staff evaluated these items as a group due to their similar function, construction, and evaluation methods. The relevant aging mechanism for these welds is IASCC. The welds are to be inspected using EVT-1 examination. The proposed analysis procedure is identical for all the items except CE-ID: 3.1, which adds some additional FMEAs for the ribs and rings. The data requirements include normal operating loads, elastic-plastic K_{app} solutions for normal operation, fast neutron fluence (or dpa) at crack location, IASCC CGR curve, limiting transient loads, K_{app} solution for limiting transient, and irradiated K_{Ic}.

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[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.

The general procedure is similar to flaw analysis procedures in the ASME Code, Section XI, Appendix C. LEFM is used for highly irradiated areas with irradiated material properties. Visually observed cracks are assumed to be through-wall. The TR indicates the calculations are expected to be plant-specific.

With respect to K_{app} solutions, the analysis procedure implied the expected flaw was a corner crack. The NRC staff requested in RAI 34, Part 4, that EPRI address the K_{app} solutions to be used if the crack is not a corner crack. In its response to RAI 34, Part 4, EPRI stated that a K_{app} solution for a through-wall flaw in a plate or shell would be the primary K_{app} solution for flaws not found at a corner. Since only the side of the core shroud weld facing the core is accessible for visual examination, in RAI 48 the NRC staff requested EPRI explain how it can be assured that cracks have not initiated on the outer diameter (OD) surface of these welds if the examination of the inner diameter (ID) surface reveals no flaws. In its response, EPRI stated that the high fracture toughness of the stainless steel assures that component integrity will be maintained for the case of a part through-wall flaw that is shorter than the allowable flaw length, and that this means there is no threat to the integrity of the structure unless the flaw propagates through the wall, in which case it would be visible from the other side.

The NRC staff reviewed the acceptance criteria and evaluation methodology and finds them to be acceptable because the acceptance criteria are based on stability of the flaw, and the evaluation methodology relies on fracture mechanics techniques that are similar in principle to those described in the ASME Code, Section XI, Appendix C. Additionally, the NRC staff finds the data requirements and inputs to the analysis are appropriate, including the use of conservative CGRs and irradiated material properties. The inputs to the analysis use conservative assumptions, such as that all visually detected flaws are through-wall. Based on the above, the NRC staff finds the acceptance criteria and evaluation methodology for the subject welds to be acceptable.

Since the TR indicates the calculations are expected to be plant-specific, as a condition on the use of the proposed methodology, the NRC staff will impose as a condition (Condition 3, Group 4) that licensees submit the plant-specific fracture mechanics analyses justifying the detected flaws 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 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

[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.

Core Shroud and Baffle Former Assemblies

- CE-ID: 4 Core Shroud Assembly (Bolted) - Assembly
- CE-ID: 5 Core Shroud Assembly (Welded) – Assembly
- W-ID: 8 Baffle-Former Assembly - Assembly

MRP-227-A requires visual examination for all these components. The relevant aging effect is distortion due to void swelling, and for W-ID: 8, cracking due to IASCC is also an aging effect. The methodology involves performing a FMEA to determine the specific acceptance criteria. For Appendix C Item CE-ID: 5 under "Failure Criteria," one of the criteria is "Gap size implies peak shroud swelling > 5% by volume." For CE-ID: 5, the proposed approach is a generic effort to support inspection, that would include an extension of the MRP model (to be discussed later) to look at the relationship between swelling and deformation at the seam, and a guideline for issues to be addressed in plant-specific FMEA. For the other two items, the TR states that the approach for this evaluation is for the FMEA to address plant-specific practices and priorities, and that some generic work is possible to outline issues and options in FMEA.

According to MRP-227-A, 5 percent swelling is considered severe and can correlate with extremely low fracture toughness values. Therefore, in RAI 40, Part 1, the NRC staff requested that EPRI discuss whether the failure criteria for gap size is correlated with a lower percentage of swelling in order to provide some margin before the onset of severe swelling. EPRI's response can be summarized as follows: Any observation of an off-normal gap would trigger evaluation under this standard. Core shroud models documented in MRP-230, "Materials Reliability Program: Functionality Analysis for Westinghouse & Combustion Engineering Representative PWR Internals," dated October 31, 2009 (proprietary report – non-publically available) indicate any severe swelling would be highly localized. The major concerns would be stresses and distortion rather than loss of ductility. Therefore, a lower limit on swelling to provide a margin before the onset of peak swelling is not needed. In its response to RAI 40, Part 2, EPRI indicated that models similar to that documented in MRP-230 would be used to correlate the observed gap to the swelling level in the material, and it is not anticipated that a full shroud model would need to be developed due to the localized nature of the swelling expected.

EPRI has previously performed FMEA work or failure modes, effects, and criticality analysis (FMECA) on the CE welded core shroud assembly. In its response to RAI 40, Part 3, which concerned the applicability of the previous work to the TR, EPRI provided criteria for applicability of the generic FMEA. These criteria are: (1) the inclusion of VT-3 inspections of reactor internals core support structures in the ASME Section XI 10-year In-service inspection program, (2) no indications or relevant condition observed to date in past inspections, (3) any pre-2007 modifications are enveloped by MRP-227-A, and more recent modifications have been evaluated and found to have no impact on the MRP-227-A aging management recommendations. EPRI also stated they would add these criteria in the final version of the TR for Appendix C Item CE-ID: 5. The staff verified this change is incorporated in the revised methodology for CE-ID: 5 provided in EPRI's January 16, 2014, letter (Ref. 5). The NRC staff finds EPRI's response to RAI 40 acceptable because the response has clarified that the failure criteria would not allow widespread severe void swelling, clarified the relationship of the plant-specific void swelling model to the generic model developed by the MRP, and clarified how plants would verify the applicability of the generic FMEA to the specific plant. The NRC staff finds the general approach of determining the acceptance criteria via an FMEA to be acceptable because the amount of acceptable swelling is not readily determined other than with respect to how it affects the function of the RVI.

For Items CE-ID:4 and W-ID:8, the NRC staff will impose the condition (Condition 3, Group 1) that the plant-specific FMEA must 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 MRP-227-A, Tables 5-2 or 5-3. For CE-ID:5, the NRC staff will impose the condition (Condition 3, Group 1) that the generic FMEA and the verification of the generic FMEA applicability 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 MRP-227-A Table 5-2.

CRGT Components (Westinghouse)

Two components associated with the CRGTs are Primary components for the Westinghouse RVI design per MRP-227, Rev. 0. These are:

- W-ID: 1 CRGT Assembly - Guide Plates(Cards)
- W-ID: 2 CRGT Assembly - Lower Flange Welds

Per MRP-227, the guide cards are subject to a VT-3 examination for wear. Industry experience has shown that the guidance holes in the guide cards can be distorted by wear. Eventually this can cause a lack of alignment that could cause an increase in control rod drop times or, as a worst case, a jam that would prevent insertion of a control rod. When wear occurs, the hole enlarges and can eventually wear through the full ligament on the inner edge of the guide card. The TR provides a methodology for predicting the remaining life of a guide card based on a correlation of the volumetric wear fraction to the visually observed remaining ligament. In RAI 25, the NRC staff requested information regarding the method of calculating the wear fraction and the margin in the life prediction equation. In response to RAI 25, EPRI stated the acceptance criteria provided in the TR were based on early results from an EPRI-sponsored effort to characterize wear rates in guide cards from Westinghouse CRGT assemblies, and that EPRI is undertaking additional work to develop more detailed guidance on managing wear-related degradation in the guide cards. EPRI further stated that upon completing the CRGT WCAP review, Section W-ID: 2 of the TR would be revised to implement the updated guidance and provide appropriate references to the technical basis. In its April 10, 2014, letter (Ref. 6), EPRI provided a revised evaluation methodology and acceptance criteria of the guide cards. The April 10, 2014, letter indicates the revised acceptance criteria are derived from WCAP-17451-P, Rev. 1, "Reactor Internals Guide Tube Wear - Westinghouse Domestic Fleet Operational Projects" (Ref. 26) and are based on ensuring wear is not severe enough to allow loss of guidance of the control rods that could prevent insertion or stepping of the rod cluster control assembly (RCCA). WCAP-17451-P was submitted to the NRC for information only by letter from the PWROG dated February 10, 2015 (Ref. 27).

The revised acceptance criteria focus on limiting the wear of a critical number of consecutive guide cards. The “Inspection Method” section of the methodology states that calibrated optical methods will be used to measure wear. This is different than the VT-3 inspection specified in MRP-227-A. Under “Coverage,” the methodology states that the coverage will be 76-87 percent depending on the reactor design, and that the inspection scope shall include at least the lower six guide cards per guide tube and, when needed, the top of the continuous guidance sheaths or C-tubes. In comparison, MRP-227-A, Table 4-3, specifies a scope of 20 percent examination of the total population of CRGT’s, with all guide cards within each selected CRGT examined

[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.

specified in MRP-227-A. The revised methodology also states that no wear measurements prior to 2015 are required and that alternate wear measurement schedules may be developed based on the guidance provided in WCAP-17451-P, Rev. 1. This initial examination schedule is not consistent with MRP-227-A, which specifies the initial inspection not later than 2 RFOs after the start of the period of extended operation. The staff notes that the change in guide card inspection requirements was communicated to the PWR licensees via letter MRP 2014-006, dated February 18, 2014 (Ref. 28). The staff also notes that approving changes to the recommendations for guide card inspections specified in MRP-227-A is outside the scope of this SE.

The methodology further states that the four innermost rodlet holes per guide card and continuous section shall be inspected. The acceptance criteria in the revised guide card methodology define three "zones," green, yellow, and red, based on the amount of wear to the innermost holes of the guide card. The amount of wear is defined in terms of a percentage of the RCCA rodlet diameter. For example, for similar amount of wear between all the guide cards in one guide tube, if the slot width is 0.8 times the rodlet diameter or less, the wear is considered to be in the green zone. Requiring the slot width to be some fraction of the rodlet diameter provides some margin for future wear. More margin is required by the proposed methodology for similar wear between all the cards because all the cards have similar wear severity close to the maximum allowed, potentially allowing the rodlet to break out at multiple cards at once. Therefore, for variable wear between cards in the same CRGT, the largest slot width is allowed to be 0.85 times the rodlet diameter for green. A larger margin of 0.75 for green is specified for the top end of the lower continuous guidance sections at which wear is most critical and rod jamming is most likely to occur. Corrective actions vary depending on whether the wear is classified as green, yellow, or red. For green, the only required action is re-inspection based on a predictive wear calculation. For yellow, corrective actions include rod drop time trending, replacement of the CRGT, engineering evaluation, and determination of operating time to avoid operating in the red zone.

The methodology did not provide sufficient detail on the basis for the margins built into the acceptance criteria, the basis for the initial inspection schedule, or the method by which the scheduling of follow-up inspections is determined. Therefore, in follow-up RAI 4, the staff requested additional details on these issues. The staff also stated in follow-up RAI 4 that EPRI may choose to provide WCAP-17451-P for information and answer each question by referring to the appropriate section in the WCAP. EPRI provided WCAP-17451-P, Rev. 1 for information to the NRC by letter dated February 10, 2015 (Ref. 28).

In Part 1 to follow-up RAI 4, the staff requested EPRI to provide the basis for the margins included in the acceptance criteria, for example, the selection of 0.8 x DR (where DR is the rodlet diameter), as the maximum allowable measured slot width for wear to be classified as "green," in the case where similar wear is observed in all the guide cards in the same control rod guide tube assembly. EPRI's response stated the response is contained in WCAP-17451-P, Subsection 5.2.1, and that the original criteria used 0.9 x DR as a "red" zone. EPRI's response further stated that one or two cycles' worth of material was added, along with additional conservatism, and that as the acceptance criteria process evolved, 0.8 x DR was settled on as the boundary for "green." The staff reviewed the information in WCAP-17451-P, Subsection 5.2.1. The staff notes that the boundaries for the green, yellow, and red zones are somewhat arbitrary. However, allowable operating times after inspection are based on [

] The staff notes that basing the zone boundaries on different fractions of the rodlet diameter is analogous to the use of structural factors in ASME Code flaw evaluations. The staff's concern in follow-up RAI 4, Part 1, is resolved since EPRI provided the basis for the margins included in the acceptance criteria.

In follow-up RAI 4, Part 2, the staff requested that EPRI provide the generic schedule and the basis for the generic schedule inspection dates, and to explain why no wear measurements are required prior to 2015. EPRI's May 5, 2015, response referred to WCAP-17451-P, Section 5.3. The staff reviewed WCAP-17451-P, Section 5.3 and finds that it provides an adequate basis for the initial inspection schedule, because [

] and thus determine the appropriate timing for initial MRP-227-A inspections. The initial inspection scheduling used the [

]. The staff notes that Enclosure 2 to EPRI Letter MRP 2014-006 (Ref. 28) also discusses the basis for the initial guide card inspection schedule, stating that determination of when initial guide tube inspection measurements should be performed is based on a review of numerous foreign material examination videos of guide tube interiors performed at many plants as part of previous guide tube support pin replacement projects and from previous guide tube wear inspections performed for the PWROG. Enclosure 2 to MRP 2014-006 further states that results of the maximum wear per plant are provided, and that with these examination results the operational extension curves are used to predict when the first inspections should be performed. Therefore, the staff's concern of follow-up RAI 4, Part 2, is resolved.

In follow-up RAI 4, Part 3, the staff requested that EPRI summarize the OE supporting the statement that the largest amounts of wear are typically observed in lowest guide card levels, and to provide a justification for only requiring inspection of the lowest six guide cards based on OE. EPRI's May 5, 2015, response referred to WCAP-17451-P, Section 2.3 and Subsection 3.5.1, stating these sections give examples of OE wear data and wear prediction models. The staff reviewed the information from OE and modeling in the referenced sections of WCAP-17451-P, and found that it adequately supports requiring inspection only of the lowest six guide cards.

In follow-up RAI 4, Part 4, the staff requested that EPRI provide the basis for the method of the predictive wear calculations that will be used to calculate the time until the next inspection for guide cards with wear in the green zone or yellow zone. EPRI's May 5, 2015, response referred to WCAP 17451-P, Section 4. The staff noted that [

] Based on its review of information in Section 3 and 4 of WCAP-17451-P, the staff finds that EPRI adequately described the method and basis that will be used to calculate the time until the next inspection for guide cards with wear in the green zone or yellow zone. The staff finds the method for the predictive wear calculations to be acceptable because it is based on [

]. The staff's concern in RAI 4, Part 4, is therefore resolved.

In follow-up RAI 4, Part 5, the staff requested EPRI provide a diagram showing "ligament wear depth." EPRI's May 5, 2015, response referred to Figure 5-1 of WCAP-17451-P. The staff reviewed this figure, and the accompanying text, which shows how the [
]. Based on its review of this information, the staff's concern in follow-up RAI 4, Part 5, is resolved.

The staff finds the evaluation methodology and acceptance criteria for the guide cards acceptable because it provides a methodology for measuring wear that is based on ensuring functionality of the RCCAs, and the acceptance criteria provide margin for future wear. In addition, the WCAP-17451-P report provides a rigorous and comprehensive basis for the methods and criteria for guide card wear evaluation.

The evaluation of the CRGT lower flange welds is similar in principle to the proposed evaluation method for bolting. Per MRP-227, Rev. 0, the welds are visually inspected using EVT-1 examination. The TR recommends that FEA be performed to determine minimum acceptable patterns of intact welds. The inspection results are compared to the minimum acceptable patterns of intact welds, with any weld with a crack like indication considered to be completely failed. The TR indicates the approach is for plant-specific analysis due to the large variety of sizes and designs, with some potential for smaller plant groupings.

The data requirements for the evaluation include the loads and the FEA model calibrated to benchmark data. Allowable loads are defined by "empirical testing." The NRC staff requested in RAI 26, Parts 1 and 2, that EPRI clarify what was meant by empirical testing and whether this was the basis for the benchmark data, and in RAI 26, Part 3, whether the loads would include LOCA and safe shutdown earthquake (SSE) loads. In its response to RAI 26, Parts 1 and 2, EPRI clarified that the empirical testing has been conducted in the past and is the basis for the benchmark data, and that no new testing will be required. In response to RAI 26, Part 3, EPRI stated that the design basis assumptions may include SSE and LOCA but are plant-specific. The NRC staff requested in RAI 26, Part 4, that EPRI discuss how it would be assured that additional welds do not have undetected cracks since the welds are only examined visually, and limited accessibility prevents inspection of 100 percent of the welds. In response to RAI 26, Part 4, EPRI indicated that the inspection strategy is based on a sampling inspection of welds that are accessible for inspection, and if no degradation is detected the entire population is considered acceptable. EPRI further indicated that if degradation is detected in individual welds, additional actions, such as expansion of the sample, would be determined by the plant corrective action program. EPRI further stated that justification is required to provide reasonable assurance that the uninspected welds are intact or that their degradation would not affect performance. The NRC staff finds the assumption that the entire uninspected population is intact based on no observed failures in the sample is not acceptable, because it is not based

on statistics. Therefore, in RAI F, Part 1, the NRC staff requested that EPRI clarify how reasonable assurance of the structural integrity of the entire population of CRGT lower flange welds will be achieved regardless of whether degradation is found in the inspected sample. In its response to RAI F, Part 1, EPRI indicated that the TR will be modified to reference NUREG-1475, Rev. 1, “Applying Statistics” (Ref. 24) with respect to justification of the sample size. NUREG-1475 is a general reference on use of statistics and provides several different statistical procedures that could be used to make inferences regarding the uninspected population.

RAI 26, Part 5, addresses the re-inspection interval if cracked welds are found and how margin for additional weld failures is addressed in the acceptance criteria. In response to RAI 26, Part 5, EPRI stated CRGTs returned to service with any flawed welds would be re-inspected after one refueling cycle, with the subsequent re-inspection frequency to be determined based on the results of the first subsequent inspection. However, the NRC staff was concerned that finding failed welds in a small number of CRGT lower flanges could represent the leading edge of a time-dependent failure distribution. Therefore, in RAI F, Part 2 the NRC staff requested EPRI discuss the need to re-inspect the entire accessible population of CRGT welds, or some sample size beyond the single defective CRGT if degradation of some welds is found in the initial inspection, and modify the TR accordingly. In response to RAI F, Part 2, EPRI noted that the CRGT lower flange welds are the “Primary” component item linked to several “Expansion” component items, and that observation of cracked CRGT lower flange welds is intended to trigger an evaluation of the extent of condition under the MRP-227-A expansion process. The NRC staff agrees that individual licensee’s quality assurance programs required by 10 CFR Part 50, Appendix B should address the extent of condition when relevant conditions are found in the CRGT lower flange welds. For additional assurance, the following paragraph will be added by EPRI to Appendix E of the TR, Item W-ID: 2:

“The inspection plan for the control rod guide tubes is a sampling approach. If indications are detected, then evaluation of the CRGT(s) with the indication must be performed. This may lead to an increase of the sample being inspected, or subsequent additional inspections.”

As modified by the responses to RAI 26 and RAI F, the NRC staff finds that the proposed acceptance criteria and evaluation methodology for the CRGT lower flange welds are acceptable. The FEA for the minimum acceptable patterns will ensure structural integrity of the CRGT lower flange weld joint. Conservative assumptions about the integrity of inaccessible welds based on statistical analysis should ensure that the as-found results are conservatively evaluated as compared to the acceptable minimum weld patterns. Further, licensee corrective action programs will ensure appropriate re-inspection schedules and scope if relevant conditions are found. Therefore, the proposed evaluation methodology will ensure sufficient welds are intact thus ensuring that the CRGT can continue to perform their function. The staff verified that the proposed revised evaluation procedure for the CRGT lower flange weld provided in EPRI’s October 31, 2014 letter (Ref. 8) incorporates the changes proposed in the responses to RAI 26 and RAI F. Therefore, the staff’s concerns in RAI 26 and RAI F are resolved. Since the TR indicates plant-specific analyses will be required, the NRC staff imposes a condition (Condition 3, Group 5) that 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.

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[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.

Instrumentation Guide Tubes

This category contains the following items:

- W-ID: 2.4 BMI System – bottom-mounted instrumentation (BMI) Column Bodies;
- CE-ID: 12 Control Element Assembly - Instrument Guide Tubes;
- CE-ID: 12.1 Remaining Instrument Guide Tubes

Although these items are located in different parts of the RVI for the Westinghouse and CE designs and have different aging mechanisms, these three items are evaluated together because the components have similar functions and are evaluated similarly. These components define the path for nuclear instrumentation (flux thimbles, in-core instrumentation) tubes to be inserted into the fuel assemblies.

Per MRP-227, Rev. 0, CE-ID: 10 Control Element Assembly - Instrument Guide Tubes, are a Primary component item to be examined via VT-3. The aging effect is cracking caused by SCC or fatigue that results in missing supports or separation at the welded joint. One hundred percent of the peripheral tubes are inspected. If degradation is found, the inspection scope is expanded to the remaining (non-peripheral) instrument guide tubes (CE-ID: 10.1). The acceptance criterion from the TR for the instrument guide tubes is that there are sufficient unfailed guide tubes such that the remaining nuclear instruments can perform the required core monitoring function. The minimum number of instrumentation guide tubes is based on the minimum number of instruments that is specified in each plant's technical specifications. No margin is required as long as the required instrumentation is functional at start up.

The NRC staff finds the acceptance criteria and evaluation method for CE-ID: 10 and CE-ID: 10.1 acceptable because it will ensure enough instruments remain operable to support plant operation per the technical specifications..

For W-ID: 2.2, BMI System - BMI Column Bodies, the aging effect is cracking due to fatigue. MRP-227-A categorizes the BMI column bodies as an Expansion component item for the CRGT lower flange welds, but also states that these components are inspected via VT-3 only in the event that difficulty of insertion or withdrawal of flux thimble tubes occurs, which occurs at the beginning and end of the RFO. Under inspection, the required coverage is described as 100 percent of BMI column bodies for which difficulty is detected during flux thimble insertion/withdrawal.

In RAI 28, Part 1, the NRC staff requested clarification of whether the BMI column bodies would be inspected if indications were found in the CRGT lower flange welds, or only if difficulty in inserting a flux thimble occurs. In its response, EPRI indicated that the BMI column bodies would be inspected only if both conditions existed (i.e., indications were found in the CRGT lower flange welds and difficulty inserting some flux thimbles occurred). However, in response to RAI 28, Part 2, EPRI indicated that it would maintain consistency with MRP-227-A with respect to the minimum percentages of expansion components to be inspected. The acceptance criterion provided in the TR is that the plant must have minimum number of functional BMI assemblies to allow flux-mapping at startup. The minimum number of functional flux thimbles is defined by the plant technical specifications. The analysis consists only on an evaluation of the stability of any failed BMI columns to determine if there is the potential to generate a loose part. The approach for the BMI column bodies is plant specific since the minimum number of instrumentation tubes is based on the plant technical specifications.

In RAI 28, Part 3, the NRC staff requested that EPRI clarify, if difficulty is encountered inserting flux thimbles at the beginning or end of RFO, whether inspection of all BMI bodies is required, and whether the inspection can be deferred until the subsequent RFO if the insertion difficulty occurs at the end of the RFO. In its response, EPRI indicated that provided there were no flawed CRGT welds that would trigger inspection of the BMI column bodies as an Expansion component item, difficulty in insertion of flux thimbles would be handled through the normal plant corrective action process, and that inspection could be deferred until the next RFO as long as a sufficient number of flux thimbles can be inserted to meet technical specification or technical requirements.

The NRC staff finds the response to RAI 28, Parts 1-3, acceptable because it clarifies when the BMI column bodies would be inspected, the minimum percentage of BMI column bodies to be inspected if expansion is triggered, and the requirements if difficulty in insertion is experienced at the end of a RFO, without CRGT lower flange weld cracking.

The NRC staff finds the acceptance criteria and evaluation methodology for the BMI column bodies and CE instrumentation guide tubes to be acceptable since they will ensure an adequate number of functional nuclear instruments and will prevent generation of loose parts.

Hold-Down Spring (Westinghouse)

This evaluation applies to the following item:

- W-ID: 7 Alignment and Interfacing Components - Internals Hold-down spring

Per MRP-227-A, the hold-down spring is a Primary component item that is inspected for stress relaxation. The inspection method is a direct measurement of the spring height. The acceptance criterion is that the relaxation of the hold down spring is above the bounding prediction, and is projected to remain so to the end of the inspection interval. Per MRP-227-A, the spring life can be extrapolated to 60 years after sufficient data on relaxation is collected (one inspection may be sufficient). The observable effect of stress relaxation is that the spring height is reduced. Repeated measurements would show reduced height from cycle to cycle.

The data requirements for the evaluations include historical information on spring height, the effective spring constant, the required hold-down force, which is plant-specific, the current spring height, and the degradation (trending). The analysis consists of construction of a creep-stress relaxation model to define bounding (high relaxation) behavior. Material properties, stiffness, creep history, geometry, and force profile are listed as analysis inputs. The acceptable value of stress relaxation is determined by plant-specific design requirements.

The NRC staff finds the analysis methodology and acceptance criteria for the hold-down spring to be acceptable because it will ensure an acceptable amount of hold-down force remains until the next inspection. The NRC staff notes that acceptance criteria related to the physical measurements performed on the hold-down spring are required to be submitted with plant-specific RVI inspection plans by Applicant/Licensee Action Item 5 of Revision 1 of the NRC staff's final SE of MRP-227, Rev. 0.

Thermal Shield Assembly (Westinghouse)

This evaluation applies to the following item:

- W-ID: 10 - Thermal Shield Assembly - Thermal Shield Flexures

Per MRP-227-A, the thermal shield flexures are a Primary component item that is inspected for cracking due to fatigue via VT-3. The acceptance criterion from the TR is that a failure of a thermal shield flexure is acceptable if it can be demonstrated that the dynamic response of the thermal shield is unchanged when the flexure is removed from the model. The TR also recommends in the case of a failed flexure that vigilance for fatigue or vibration-monitoring systems should be enhanced. Any thermal shield flexure with an observed flaw is assumed to be failed, and no credit is taken for “bumpers” and other redundant structures. The analysis consists of a structural assessment to determine the minimum number of flexures required to retain structural integrity. The structural analysis will establish the dynamic response of the thermal shield.

In RAI 44, the NRC staff requested additional information concerning what recommendations would be made if the dynamic response of the thermal shield did change as a result of the failed flexures, and more detail on how the potential for fatigue should be monitored if failures were found. In its response to the first part of RAI 44, EPRI stated that 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 interval for re-inspection. In its response to follow-up RAI G, EPRI confirmed this guidance would be added to the final version of the TR. In response to the second part of RAI 44, EPRI stated that enhanced vigilance could involve several approaches, such as more frequent inspection during outages, loose parts monitoring, and neutron noise monitoring. The staff confirmed the guidance described in the response to RAI 44 has been incorporated in the proposed revision of the methodology and acceptance criteria for the thermal shield flexures provided in EPRI’s April 10, 2014, letter. RAI 44 and RAI G are thus resolved.

The NRC staff finds the proposed evaluation methodology and acceptance criteria for the thermal shield flexures to be acceptable because the criteria will ensure the thermal shield continues to have adequate structural support and stability such that it can continue to perform its function. Since the TR indicates plant-specific analyses will be required, the NRC staff imposes a condition (Condition 3, Group 5) that 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.

Other Core Support Structures

- 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

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[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.

Since Condition 1 of the staff's final SE of MRP-227, Rev. 0 required the Upper Internals Assembly – Upper Core Plate and Lower Internals Assembly – Lower Support Forging or Casting to be moved from the No Additional Measures category to the Expansion category, in RAI 45, the staff asked EPRI to confirm that evaluation procedures for these components would be added to the TR. EPRI's April 10, 2014, letter (Ref. 6) contained the evaluation procedures for these components. As it committed in its response to RAI 36, EPRI provided a revised evaluation procedure for the Upper Internals Assembly – Fuel Alignment Plate in its January 16, 2014, letter (Ref. 5), which makes the evaluation procedure for this component consistent with the Lower Support Structure – Core Support Plate. The staff's concerns in RAI 36 and RAI 45 are thus resolved.

These components are all categorized as Expansion and would be subject to EVT-1 visual inspection if expansion was triggered. All of these components are in the form of a flat, circular plate with numerous holes. The evaluation procedures include the assumption that multiple cracked ligaments between holes would have to exist to degrade the functionality of the component. Detected cracks are assumed to initiate at the location in the component with the highest tensile stress and grow completely through-wall and through the ligament between adjacent holes. An FEA of the component is performed. Acceptance criteria are based on maintaining the deflection or displacement and stresses of the plate within acceptable limits. The methodology does not include CGRs or fracture toughness properties. If acceptability of a single crack through the entire ligament cannot be demonstrated, a detailed flaw analysis is required. Re-inspection is required during each RFO unless additional analysis can justify a longer re-inspection interval.

The staff finds the evaluation procedures for these components to be acceptable because the assumption that any crack initiates at the high-stress location and is completely through the ligament is conservative, and because the acceptance criteria are based on functionality of the components. However, since the TR indicates plant-specific analyses will be required, the NRC staff imposes as a condition (Condition 3, Group 6) that the plant-specific analyses must be submitted for the NRC to determine if review and approval is needed, within one year after any inspection that detects relevant conditions as defined in MRP-227-A, Table 5-2, for the following RVI components:

- 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

4.0 CONDITIONS

Based on its review, the NRC staff identified some issues and concerns in Section 3.0 of this SE regarding the implementation of the 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 the inspection results do not meet the acceptance criteria or trigger the expansion criteria methodology is used. Conditions are imposed to ensure submittal of these analyses. It should be noted, however, that any limitations and conditions specified in this SE are considered as supplements (not replacements) to the limitations and conditions specified in the SE dated December 16, 2011, contained in the MRP-227-A report.

4.1 B&W RVI Components

<u>Condition 1</u>

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.

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 5-1 of MRP-227-A.

4.2 B&W, CE, and Westinghouse RVI Components

<u>Condition 2 – Crack Growth Rates for High-Fluence Components</u>

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.

4.3 CE and Westinghouse RVI Components

<u>Condition 3</u>

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

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[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.

For Group 2 RVI Component Items:

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:

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[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 not resubmit plant-specific analyses unless the inspection findings are not bounded by the original analysis.

- 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

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

[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.

- 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

5.0 CONCLUSIONS

The NRC staff has reviewed the TR, and concludes that the report, as modified by the conditions detailed in Section 4.0 of this SE, describes acceptable methods for evaluating relevant conditions and assessing aging degradation that may be found in RVI by licensees conducting inspections in accordance with MRP-227-A. Upon incorporation of the changes to which EPRI committed in various RAI responses, and the revised evaluation procedures that were provided in its January 16, April 10, and July 8, 2014, letters, the NRC staff concludes the TR is acceptable for referencing in plant-specific RVI AMPs and RVI inspection plans that may be submitted in conjunction with or subsequent to an application for a renewed license.

Before endorsement by the NRC, the TR must be updated to reflect the incorporation of the changes to which EPRI committed in various RAI responses, and the revised evaluation procedures that were provided in its January 16, April 10, and July 8, 2014, letters.

6.0 REFERENCES

1. Letter from Terry McAlister to NRC dated May 19, 2010; Subject: Report Transmittal, Westinghouse Non-Proprietary Class 3 Report WCAP-17096-NP, Revision 2, "Reactor Internals Acceptance Criteria Methodology and Data Requirements," December 2009; MRP Letter No. 2010-034 (ADAMS Accession No. ML1014601560).

2. Letter from Terry McAlister to NRC, Revised Responses to the NRC Request for Additional Information (RAI) on PWR Owners Group (PWROG) WCAP-17096-NP, Revision 2, "Reactor Internals Acceptance Criteria Methodology and Date Requirements" (TAC No. ME4200) PA-MSC-0473, June 14, 2012 (ADAMS Accession No. ML12171A374).

3. Letter from Terry McAlister to NRC, Revised Responses (LTR-RIAM-12-138, Rev 1) to the NRC Request for Additional Information (RAI) on PWR Owners Group (PWROG) WCAP-17096-NP, Revision 2, "Reactor Internals Acceptance Criteria Methodology and Data Requirements," TAC No. ME4200, PA-MSC-0473, January 3, 2013 (ADAMS Accession No. ML13008A161).

4. Letter from Timothy Wells to NRC, Proposed Edits to WCAP-17096-A Draft, August 5, 2013 (ADAMS Accession No. ML13219A183).

5. Letter from Timothy Wells to NRC, Proposed Edits to WCAP-17096-A Draft, January 16, 2014 (ADAMS Accession No. ML14041A042).

6. Letter from B. C. Rudell to NRC, Proposed Edits to WCAP 17096-NP, Revision 2, April 10, 2014 (ADAMS Accession No. ML14104B579)

7. Letter from B. C. Rudell to NRC, Transmittal of Westinghouse Revised Text for Draft WCAP-17096-NP Revision 2 for NRC Review, LTR-RIAM-12-159, Revision 2 (PA-MSC-0473R5), July 8, 2014 (ADAMS Accession No. ML14191A014)

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

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

10. Westinghouse Non-Proprietary Class 3 Report WCAP-17096-NP, Revision 2, “Reactor Internals Acceptance Criteria Methodology and Data Requirements,” December 2009 (ADAMS Accession No. ML101460157).

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12. MRP-191, “Materials Reliability Program: Screening, Categorization and Ranking of Reactor Internals of Westinghouse and Combustion Engineering PWR Designs,” November 2006 (ADAMS Accession No. ML091910130).

13. Carter, Robert and Pathania, Raj, “Technical Basis for BWRVIP Stainless Steel Crack Growth Correlations in BWRs,” PVP2007-26618, ASME Pressure Vessel & Piping Conference Proceedings, July 2007.

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18. BWRVIP-99-A, “BWR Vessel and Internals Project Crack Growth Rates in Irradiated Stainless Steels in BWR Internal Components,” Final Report, November 2008.

19. BWRVIP-76-A, “BWR Vessel and Internals Project, BWR Core Shroud Inspection and Flaw Evaluation Guidelines,” December 31, 2009 (ADAMS Accession No. ML101530467).

20. J.K. McKinley, et al., “CGR and Fracture Toughness of Austenitic Stainless Steel in a PWR Primary Water Environment,” 14th International Conference on Environmental Degradation of Materials in Nuclear Power Systems, August 23-27, 2009.

21. WCAP-15030-NP-A report, “Westinghouse Methodology for Evaluating the Acceptability of Baffle-Former-Barrel Bolting Distributions under Faulted Load Conditions,” March 2, 1999 (ADAMS Legacy Library Accession No. 9905070171)

22. Westinghouse Proprietary Report WCAP-15029-P-A, Rev. 1, “Westinghouse Methodology for Evaluating Acceptability of Baffle-Former-Barrel Bolting Distributions Under Faulted Load Conditions,” December 31, 1998 (ADAMS Legacy Library Accession No. 9905070164).

23. MRP-175, “Materials Reliability Program: PWR Internals Material Aging Degradation Mechanism Screening and Threshold Values,” December 2005 (ADAMS Accession No. ML063470637).

24. NUREG-1475, Rev. 1, “Applying Statistics,” March 2011 (ADAMS Accession No. ML11102A076)

25. Letter from Joseph Holonich to B. C. Rudell, Request for Additional Information Related to WCAP-17096-NP, Revision 2, "Reactor Internals Acceptance Criteria Methodology and Data Requirements" (TAC No. ME4200), August 18, 2014 (ADAMS Accession No. ML14177A071)

26. WCAP-17451-P, Rev. 1, "Reactor Internals Guide Tube Wear - Westinghouse Domestic Fleet Operational Projects," October 2013 (ADAMS Accession No. ML15041A107) – PROPRIETARY – NOT PUBLICALLY AVAILABLE

27. Letter from Jack Stringfellow to NRC, PWR Owners Group, Submittal of WCAP-17451-P, Revision 1, "Reactor Internals Guide Tube Wear-Westinghouse Domestic Fleet Operational Projections," to NRC for Information Only (PA-MSC-0688), February 10, 2015 (ADAMS Accession No. ML15041A106)

28. Electric Power Research Institute Letter MRP 2014-006, Materials Reliability Program: Pressurized Water Reactor Internals Inspection and Evaluation Guidelines (MRP-227-A), February 18, 2014 (ADAMS Accession No. ML14274A372)

Attachment: Staff Resolution of MRP Comments on Draft Safety Evaluation

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Date: May 3, 2016

WCAP-17096-NP, Rev. 2 – Staff Resolution of MRP Comments on Draft Safety Evaluation

Comment #	Page #	Section #	Paragraph / Sentence #	Table #	Comment	Resolution
1	3	3.1	2nd paragraph, last sentence		Reference to TLAA has been removed from WCAP-17096 via RAI responses	Guidance for TLAAs was removed from evaluation procedures via RAI 3 response dated May 5, 2015 (ML15128A199). The RAI response did not include a markup of Section 3.6 of the TR which discusses the components resolved by TLAA. The following sentence was therefore added to the paragraph: "However, in the response to follow-up RAI 3 dated May 5, 2015 (Ref. 9), EPRI clarified that guidance for TLAAs is not within the scope of WCAP-17096-NP." The staff recommends EPRI consider modifying Section 3.6 of the TR to reflect the RAI response.
2	5 (and several others)	3.2.1	1st paragraph, last sentence		The text "within one year of the initial inspection…" can be interpreted multiple ways. One interpretation could be that a licensee need not submit an analysis should any future examination results indicate aging that triggers expansion. It is suggested that wording similar to that used in Section 3.2.2.1 for the CRGT and IMI guide tube spiders and welds be used if the intention of the statement is the same - "...submit, within one year of inspection of the IMI guide tube assembly spiders or CRGT spacer castings, 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." This comment applies to the other locations where this wording is	Removed the word "initial" before "inspection" in the condition wording.

Comment #	Page #	Section #	Paragraph / Sentence #	Table #	Comment	Resolution
					used throughout the document (e.g., Section 3.2.2.1 and Section 4.1)	
3	5	3.2.1	1st paragraph, 4th sentence		It is believed that the "P" in brackets is simply the staff adjusting content as stated in WCAP-17096. This also occurs on Page 8 with "PH" in brackets. Please clarify what is meant by the use of these brackets in these instances.	The brackets indicate changes from the exact quote in the source document. In the first case the staff capitalized “primary” where it was not capitalized in EPRI’s response. In the second case “precipitation-hardeneable” was spelled out in the TR text and the staff abbreviated PH.
4	6-7	3.2.2.1	Last paragraph, last sentence		Wording on the bottom of page 6 refers to the FD bolts in the second bullet on page7 as being an "expansion" item, but as discussed in RAI responses, the WCAP is to be revised to be consistent with FD bolts being a "Primary" item in MRP-227-A, per the SE for MRP-227-A. Thus, minor modification to the wording is required in order for this draft SE to be accurate.	Change made, additional bullet added for FD bolt and referred to Primary components in the first two bullets in the condition wording.
5	25	3.2.2.2.2	2nd paragraph, last sentence		There are no expansion criteria for the baffle-edge bolts.	Agreed. Staff revised wording of condition to read “The NRC staff will impose a condition (Condition 3, Group 1) for plant-specific application of this TR that any generic or plant-specific FMEA methodology be submitted to the NRC to determine whether review and approval is

Comment #	Page #	Section #	Paragraph / Sentence #	Table #	Comment	Resolution
						needed, within one year after inspection results detect relevant conditions for the baffle-edge bolts as defined in the "Examination Acceptance Criteria" column of Table 5-3 of MRP-227-A."
6	32 (and other locations)	3.2.2.2.2	3rd paragraph, 3rd sentence		For several of the conditions imposed by the draft SE, the following text or similar text appears: "…that plant-specific analyses be submitted to the NRC to determine if review and approval is needed, within one year after inspection results indicate aging triggering the expansion criteria of MRP-227-A Table 5-2 or 5-3." Similar text appears in the following locations: - Page 32, Condition 3, Group 3 - Page 33, Condition 3, Group 4 - Page 34, Condition 3, Group 1 - Page 39, Condition 3, Group 4 - Page 42, Condition 3, Group 4 - Page 42, Condition 3, Group 6 - Page 44, Condition 3 section The requirements of the condition are currently unclear. Specifically, the components the conditions are applicable to; the submittal timing for each class of component; and the intention to use the expansion criteria when determining if an analysis must be submitted.	The wording of the conditions has been modified to clarify the type of analysis to be provided, and to clarify exactly which components to which each condition is applicable. Additionally, a footnote has been added to each condition clarifying that the one-year time frame begins on the date the plant begins power operation upon startup from the refueling outage during which the degradation was discovered. In addition, the previous language in the draft SE that required the analyses to be submitted only when the expansion criteria was triggered has been modified because most Expansion components and some Primary components do not have expansion criteria. For components that do not have expansion criteria, conditions were thus modified so that the analysis must be submitted within one year of any inspection detecting apply when the relevant conditions as defined in MRP-227-A Tables 5-2 or 5-3, or the acceptance criteria defined in these tables are exceeded if quantitative acceptance criteria exist (for example, if UT examinations are specified).. For Primary components, the relevant

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						conditions or acceptance criteria are contained in the "Examination Acceptance Criteria" column of MRP-227-A Table 5-2 (for CE components) or 5-3 (for Westinghouse components), and for Expansion components, the relevant conditions or acceptance criteria are contained in the "Additional Examination Acceptance Criteria" column of the same tables. For components that have expansion criteria, the conditions are unchanged, i.e., the analysis still does not have to be submitted unless the inspection results trigger the expansion criteria.
7	34	3.2.2.2.2	2nd paragraph, 3rd sentence		Consider revising the sentence starting with "These criteria are the inclusion of VT-3 inspections of…" to avoid future potential for misunderstanding.	Added numbering to items listed in sentence.
8	34	3.2.2.2.2	3rd paragraph, 1st sentence		CE-ID: 4 and W-ID:8 are primary components with no expansion links	Modified sentence as follows: For Items CE-ID:4 and W-ID:8, the NRC staff will impose the condition (Condition 3, Group 1) that the plant-specific FMEA must be submitted for the NRC to determine if review and approval is needed, within one year after inspection results indicate relevant conditions as defined in the "Examination Acceptance Criteria" column of MRP-227-A, Tables 5-2 and 5-3.

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9	43	4.1	3rd paragraph		The paragraph starting with "As discussed in Section 3.2.1 of this SE,…" is a long sentence and unclear as is the paragraph starting with "As discussed in Section 3.2.2.1 of this SE,…" two paragraphs later. Please consider revising for clarity.	The paragraphs are modified to avoid misinterpretation.
10	43	4.1	3rd paragraph		It appears that an inconsistency exists between the MRP-227-A SE and this draft SE of WCAP-17096 in Condition 1. A/LAI #6 in the MRP-227-A SE states the following: "Applicants/licensees shall justify the acceptability of these components for continued operation through the period of extended operation by performing an evaluation, or by proposing a scheduled replacement of the components. As part of their application to implement the approved version of MRP-227, applicants/licensees shall provide their justification for the continued operability of each of the inaccessible components and, if necessary, provide their plan for the replacement of the components for NRC review and approval." Section 4.1 of this draft SE states the following: "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 initial inspection of the linked Primary component items for NRC to determine	Condition 1/Group 1 in the draft SE for WCAP-17096-NP represents a relaxation of A/LAI #6 in the MRP-227-A SE, considering that in practice the applicants for B&W plants were not able to provide what A/LAI #6 requested as part of their applications to implement MRP-227-A. For future applications implementing MRP-227-A, the plant-specific inspection plan can use Condition 1/Group 1 in the draft SE for WCAP-17096-NP, instead of the more restrictive A/LAI #6 in the MRP-227-A SE. Therefore, in this case, Condition 1/Group 1 is a replacement of A/LAI #6. The following revision on Page 5 of the draft SE for WCAP-17096-NP clarifies this point: "….are specified in the December 16, 2011, SE in MRP-227-A. For future applications implementing MRP-227-A, the plant-specific inspection plan can use Condition 1/Group 1 in the draft SE for WCAP-17096-NP, instead of the more restrictive Applicant/Licensee Action Item 6 in the MRP-227-A SE. Therefore, in this case, Condition1/Group 1 is a

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					whether review and approval are needed..." Additionally, in Section 3.2.1 (Page 4) the language supplements (not replacements) is used. It is not clear what is meant by "supplements (not replacements)" nor how licensees are to adhere to the limitations and conditions in both the MRP-227-A and WCAP-17096 SEs.	replacement of Applicant/Licensee Action Item 6 in the MRP-227-A SE."
11	44	4.1	1st paragraph		The paragraph starting with "As discussed under 'CRGT Spacer Castings' and…" has some issues with clarity due to the long sentences and how the modifiers in the different clauses of the sentences could be interpreted in different ways. Please consider revising for clarity. The same comment applies to the sentence beginning with "As discussed under Section 3.2.2.2.2 of this SE, the applicant is required..." on the same page.	The paragraph is modified to avoid misinterpretation. The paragraph is not revised because misinterpretation is unlikely.

Comment #	Page #	Section #	Paragraph / Sentence #	Table #	Comment	Resolution
12	44-46	4.3	Several Locations		There appear to be inconsistencies with the component ID nomenclature as identified in Condition 3 of the draft SE. For example, some of the component IDs are consistent with that of the RAI response letters submitted on the docket, while several are not. The component IDs have been updated as a result of letters submitted on the docket in response to RAIs. It is recommended that component ID nomenclature and numbering be revised as submitted in the RAI response letters to ensure consistency.	The component nomenclature for all Westinghouse and CE components in the SE has been updated as necessary for consistency with the updated component nomenclature in the letters from MRP to NRC dated January 16, 2014 (Ref. 5), April 10, 2014 (Ref. 6), and July 8, 2014 (Ref. 7).