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

April 25, 2019

Brian Burgos
MRP Program Manager
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3420 Hillview Avenue
Palo Alto, CA 94304

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

Dear Mr. Burgos:

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

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

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

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

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

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

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

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

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

Sincerely,

/RA/

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

Docket No. 99902021

Enclosure:
Final SE

SUBJECT: FINAL SAFETY EVALUATION FOR ELECTRIC POWER RESEARCH INSTITUTE TOPICAL REPORT MRP-227, REVISION 1, "MATERIALS RELIABILITY PROGRAM: PRESSURIZED WATER REACTOR INTERNALS INSPECTION AND EVALUATIONS GUIDELINE" (CAC NO.MF7223; EPID L-2016-TOP-0001) DATE: APRIL 25, 2019

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FINAL SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

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

1.0 INTRODUCTION

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

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

2.0 REGULATORY EVALUATION

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

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

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

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

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

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

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

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

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

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

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

Additional Guidance Used or Referenced in Subsequent License Renewal Applications

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

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

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

3.0 TECHNICAL EVALUATION

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

3.1.1 Section 1 Changes

The following wording was added to the executive summary:

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

The NRC staff finds this change acceptable.

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

examination acceptance criteria defined in Section 5 of these guidelines shall be recorded and entered in the owner's plant corrective action program and dispositioned." Section 7.5 further states that engineering evaluations used to disposition an examination result that does not meet the examination acceptance criteria in Section 5, shall be conducted in accordance with NRC accepted evaluation methods (i.e., ASME Code Section XI, WCAP-17096-NP or equivalent method).

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

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

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

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

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

3.1.3 Specific Line Item Changes

3.1.3.1 Babcock and Wilcox Primary Components – Table 4-1

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

3.1.3.2 B&W expansion components

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

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

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

a. Clarify whether the expansion link column or Note 8 is correct.

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

3.1.3.3 B&W Plants Examination and Acceptance Criteria

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

3.1.3.4 Combustion Engineering Primary Components

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

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

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

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

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

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

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

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

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

In MRP-227, Revision 1, Table 4-2, three Combustion Engineering (CE) primary components state under "Examination Method/Frequency," "If screening for fatigue cannot be satisfied by plant-specific evaluation, enhanced visual (EVT-1) examination, no later than 2 refueling outages from the beginning of the license renewal period. Subsequent examination on a ten-year interval." The language for the corresponding components in MRP-227-A for "Examination scope/frequency" stated "If fatigue life cannot be demonstrated by time-limited aging analysis (TLAA), enhanced visual (EVT-1) examination, no later than 2 refueling outages from the beginning of the license renewal period. Subsequent examination on a ten-year interval."

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

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

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

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

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

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

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

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

3.1.3.5 *CE Expansion Components*

<u>CE Core Shroud Welds (expansion)</u>

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

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

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

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

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

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

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

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

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

3.1.3.6 CE Plants Examination and Acceptance Criteria

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

3.1.3.7 Westinghouse Primary Components

<u>Internals Hold Down Spring</u>

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

Baffle-Former Bolts

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

Enclosure 1 to MRP Letter 2014-006 states that:

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

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

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

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

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

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

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

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

Changes to Expansion Links

For the core barrel assembly –LGW, the upper core plate and lower support column (LSC) bodies (cast and non-cast) have been added as expansion links, in addition to the middle axial welds and lower axial welds. The axial weld expansion links are similar to the expansion link in MRP-227-A for the equivalent component, the upper and lower core barrel cylinder girth welds, for which the expansion link was the upper and lower core barrel cylinder axial welds.

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

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

3.1.3.8 Westinghouse Expansion Components

Coverage and Method Changes for Westinghouse Expansion Components

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

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

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

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

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

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

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

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

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

3.1.3.9 Westinghouse/CE Primary Components

Inspection Strategy for Westinghouse/CE Core Barrel Welds Sampling

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

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

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

In its September 28, 2018, response to RAI 29 (Ref. 6), EPRI provided markups of MRP-227, Revision 1, Table 4-2, "CE Plants primary Components," and Table 4-3, "Westinghouse Plants primary Components," that modified the proposed examination coverage of the core barrel welds as shown in Table 1. EPRI also provided markups of MRP-227, Revision 1, Table 4-5, "CE Plants expansion Components," and Table 4-6, "Westinghouse Plants expansion Components," which modified the examination coverage of the associated axial weld expansion components. For the CE primary welds, the table markup includes Note 5 stating that examination coverage requires a minimum of 75% of the length of either the ID or the OD of the weld being examined.

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

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

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

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

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

EPRI also summarized the results of tests of control rod insertability that simulated a full core drop type accident. EPRI stated that core drop testing also tested the effect of significant fuel deflections, in which the center of the fuel assembly was deflected laterally while the top and bottom were pinned. EPRI determined the effect on scram time was acceptable. EPRI proposed that this testing provides evidence that the small "bend" in the control rod insertion path that could be caused by a tilted core barrel would not impact the ability to insert the control

rods for core shutdown. In a letter dated May 17, 2018, EPRI provided additional details of this testing.

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

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

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

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

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

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

The staff finds the functionality information presented by EPRI increases the staff’s confidence that safe shutdown could be achieved even with a complete or partial girth weld failure, due to the design features in the RVI that maintain alignment and support for the core barrel. This would also apply to a full core barrel weld failure during normal operation, although EPRI did not cite the design features for a weld failure during normal operation. However, the NRC staff

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

considers it much less likely that a full core barrel weld failure would occur under normal operating conditions.

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

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

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

Reclassification of Core Barrel Welds

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

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

Table	MRP-227-A primary Item	MRP-227, Revision 1 primary Item(s)	MRP-227, Revision 1 expansion Item(s)	MRP-227, Revision 1 primary link for expansion Item(s)
4.3	Core Barrel Assembly – Upper and Lower Core Barrel Cylinder Girth Welds	Core Barrel Assembly W4. Lower Girth Weld (LGW) (primary)	Core Barrel Assembly W3.1. Upper Girth Weld (expansion) (UGW)	Core Barrel Assembly W3 Upper Flange Weld (UFW)
4.3	Core Barrel Assembly – Lower Core Barrel	n/a	Core Barrel Assembly W3.3. Lower Flange Weld	Core Barrel Assembly W3 Upper Flange

Table	MRP-227-A primary Item	MRP-227, Revision 1 primary Item(s)	MRP-227, Revision 1 expansion Item(s)	MRP-227, Revision 1 primary link for expansion Item(s)
	Flange Weld		(LFW)	Weld (UFW)

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

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

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

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

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

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

EPRI stated that, provided with this revised information on applicable degradation mechanisms and with the updated nomenclature for each of the welds, the core barrel welds were assigned a more logical structure in MRP-227, Revision 1, using its sampling and lead component strategies. The UGW was made an expansion component to the UFW because both welds

have similar low normal operating stresses and were screened in for the same degradation mechanism (SCC). The UFW has the added potential of elevated bending stresses due to the proximity of the upper flange. If the sampling inspection of the UFW as a primary lead component detects evidence of cracking degradation, the UGW will require expansion inspection according to the requirements of MRP-227, Revision 1, to monitor for further occurrence of the degradation mechanism.

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

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

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

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

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

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

The NRC staff finds the EPRI response to RAI 26 Item c to be acceptable because the staff finds the LFW should have lower stresses than the UFW, based on not experiencing the higher bending stresses experienced by the UFW. The NRC staff therefore finds it acceptable for the LFW to be reclassified as an expansion component. The staff also finds that EPRI provided a reasonable explanation for the CE CSBFW remaining a primary component while the LFW was moved to expansion. The NRC staff also notes that the UGW, UAW, and LFW screened in for SCC, but not for IASCC. SCC has rarely occurred in austenitic stainless steel in PWR reactor

coolant other than creviced or stagnant locations, which the core barrel welds are not, due to the water chemistry control in PWR water which minimizes oxidants and contaminants. Therefore, although possible, the NRC staff considers SCC a low-probability mechanism, which also supports these welds being categorized as expansion.

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

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

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

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

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

<u>CE & Westinghouse Core Support Columns</u>

In Table 4-2, CE Plants primary Components, Item C8, “Lower Support Structure – Core Support Columns,” is a new item that includes both core support columns (for plants with full height bolted core shroud plates) and core support column welds (for plants with half-height welded core shroud plates). The examination coverage for the core support columns is 25 percent of the column assemblies as visible using a VT-3 examination from above the lower core plate, and for the core support column welds is 100 percent of the accessible

surfaces. In MRP-227-A the equivalent item included only the core support column welds, with an examination coverage of 100 percent of the accessible surfaces, for all plants. In MRP-227, Revision 1, there are differences in required examination coverage for the core support column components for the two plant design variations. In addition, the component in Westinghouse-design RVI with the same function is an expansion component whereas the CE core support columns are a primary component.

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

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

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

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

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

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

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

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

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

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

EPRI also provided several factors supporting the assertion that full section cracking of LSCs is unlikely, including: quality controls during fabrication, low ferrite content such that TE or TE plus IE are not expected; low tensile stress; no likely mechanisms for flaw initiation and growth, and sufficient toughness to withstand any secondary cracking in a column that is already cracked through the cross section. In particular, the staff notes that fabrication quality controls included

liquid penetrant (PT) examination and radiography, which provides reasonable assurance that before installation the LSCs welds were structurally sound. With respect to RAI 9 Item b, use of VT-3 is adequate because it can detect fully fractured, misaligned, or missing columns, which are the only condition that can affect functionality (e.g., partially cracked columns are still functional).

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

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

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

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

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

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

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

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

The NRC staff finds EPRI's response to RAI 9 acceptable because it provides a technical justification based on maintaining functionality of the LSC and CSC for the reduction in coverage and the change in examination method for these components, based on a technical report the NRC staff has reviewed and found acceptable. Specifically, in its assessment of PWROG-14048-P, the NRC staff found the report's conclusions to be acceptable, i.e., that full

section cracking of lower support columns is extremely unlikely, and that there is significant redundancy in the lower support system (LSS) designs, such that, in the most unlikely event where complete loss of column support does occur at certain locations, the remaining intact structure will sufficiently support limiting loading conditions.

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

3.1.3.10 CE & Westinghouse Existing Programs Components

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

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

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

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

Clevis insert bolts are fabricated for Alloy X-750, a nickel-based alloy which is susceptible to primary water stress corrosion cracking (PWSCC). The plant that experienced failures detected evidence of the failures of some of the failed bolts visually. Additional detail on the failures in available in PWROG presentation slides, "Industry and NRC Coordination Meeting Materials Programs Technical Exchange: Clevis Insert Bolt Update" (June 2014) (Ref. 47), which indicates that 29 of a total of 48 clevis insert bolts were failed, and that metallurgical examination confirmed that the failures were caused by PWSCC.

Reference 47 indicated that all designs were inherently safe, and that the impact of clevis insert bolt failures is primarily commercial in nature. Reference 47 also indicates that VT-3 visual examination is an appropriate examination method for clevis insert bolts because it can monitor

conditions that affect the functional performance of the LRSS such as excessive or abnormal wear or looseness or dislocation of the clevis insert, and can also reduce the commercial risk related to clevis insert bolt failure by detecting evidence of bolt failure.

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

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

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

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

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

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

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

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

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

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

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

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

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

3.1.4 Changes to Evaluation Methodologies (TR Section 6)

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

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

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

3.2 Incorporation of Operating Experience

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

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

3.2.1 B&W Components

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

3.2.2 Westinghouse/CE Components

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

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

EPRI stated that other than the core barrel welds, to date OE has not suggested other changes that are needed to modify or clarify examination coverage requirements for other MRP-227-A components for Westinghouse plants based on the actual accessibility achieved during the examinations that have been completed.

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

accessibility concerns with these components, the inspection requirements were not changed as a result of OE.

Summary – RAI 13

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

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

3.3 Examination Coverage of Adjacent Base Metal for Welds

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

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

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

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

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

The NRC staff finds that EPRI has adequately addressed RAI 20 by clarifying that the required coverage of base metal adjacent to welds is generally ¾ of an inch on either side of the weld, and that this is a requirement of MRP-227 Revision 1 rather than just a recommendation. The staff considers that ¾ of an inch is sufficient to encompass the HAZ. Also, EPRI's addition of the ¾ inch examination coverage to Tables 4-2 through 4-6 further ensures that the coverage requirement is clear. RAI 20 is thus resolved.

3.4 Control Rod Drive Mechanism Thermal Sleeve Wear Issue

During the May 2018 Materials Information Exchange Meeting, the Electric Power Research Institute, Materials Reliability Program (EPRI-MRP) and the PWROG Materials Subcommittee

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

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

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

3. 5 Implementation Requirements

MRP-227, Revision 1, Section 7 contains the implementation requirements with respect to the NEI 03-08 protocol, which allows industry issue programs such as the EPRI MRP to designate industry guidance documents or elements as mandatory, needed, or good practice. NEI 03-08 is an industry-controlled program for managing materials integrity in nuclear power plants. As such, the NRC staff does not endorse the NEI 03-08 document, or the classification of guidance elements under it.

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

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

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

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

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

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

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

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

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

RAI 27 is thus resolved.

With respect to timing of the implementation of the needed inspections, Section 7.3 states that for units that have submitted an AMP to the regulator under [referencing] MRP-227-A, and their period of extended operation begins no later than six years from the issuance date of this guideline, that MRP-227-A based program may be implemented as the baseline inspection without deviation from this "Needed" requirement, but the program should also include the requirements contained within the interim guidance letters MRP-2014-006 and MRP-2013-023. However, subsequent implementation[5] shall be in accordance with the revision of guidelines [MRP-227] in effect at the time.

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

Section 7.3 states that if the above exception for submitted AMPs does not apply, a time-period of approximately 36 months (3 years) from the effective date of this revision is considered

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

reasonable for adopting these guidelines. Engineering program updates may be deferred until 24 months (2 years) prior to the beginning of the next inspections governed by these guidelines. The issue date of MRP-227, Revision 1 was October 2015, so plants with initial inspection scheduled to occur through October 2021 may opt to perform these inspections in accordance with MRP-227-A. Subsequent inspections for these plants could potentially be in accordance with a later revision of MRP-227.

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

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

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

A/LAI 1 stated:

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

A/LAI 2 stated that:

> Consistent with the requirements addressed in 10 CFR 54.4, each applicant/licensee is responsible for identifying which RVI components are within the scope of LR for its facility. Applicants/licensees shall review the information in Tables 4-1 and 4-2 in MRP-189, Revision 1, and Tables 4-4 and 4-5 in MRP-191 and identify whether these tables contain all of the RVI components that are within the scope of LR for their facilities in accordance with 10 CFR 54.4. If the tables do not identify all the RVI components that are within the scope of LR for its facility, the applicant or licensee shall identify the missing component(s) and propose any necessary modifications to the program defined in MRP-227, as modified by this SE, when submitting its plant-specific AMP. The AMP shall provide assurance that the effects of aging on the missing component(s) will be managed for the period of extended operation.

The intent of A/LAI 2 in the staff's final SE of MRP-227, Revision 0, was to ensure applicants and licensees identify any plant-specific RVI components that were not addressed by the generic screening, FMECA, functionality, and aging management recommendations of MRP-

227-A. This also extends to components that may have the same configuration, but are fabricated from different materials such that they may be subject to different aging mechanisms or effects not identified for the generic component in MRP-227-A.

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

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

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

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

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

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

A/LAI 3 stated:

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

A/LAI 3 was included in Ref. 16 because MRP-227-A did not provide adequate guidance for applicants/licensees to document the details of the plant-specific existing programs in plant-specific RVI programs. The NRC staff notes that MRP-227-A did state, with regard to existing plant-specific programs for CE-design RVI. that "the guidance for in-core instrumentation (ICI) thimble tubes and thermal shield positioning pins is limited to plant-specific recommendations and thus have no generic reference, nor are they included in Table 4-8. The owner should review their specific design, upgrade status, and plant commitments for CE ICI thimble tubes."

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

> The guidance for guide tube support pins (split pins) is limited to plant-specific recommendations and thus have no generic reference. Subsequent performance monitoring should follow the supplier recommendations. Thus, they are not included in

Table 4-9. The owner should review their specific design, upgrade status, and asset management plans for guide tube support pins (split pins).

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

Therefore, in RAI 15 the staff requested EPRI:

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

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

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

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

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

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

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

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

A/LAI 4 stated that:

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

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

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

A/LAI 5 stated:

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

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

The NRC staff issued RAIs to some applicants/licensees of Westinghouse-design plants asking for a description of the methodology for determining the hold-down spring height acceptance criteria. The methodology described in all these responses assumes a linear decrease in spring height from the preservice measurement to the required spring height at end of life (60 years) with the acceptance criteria at any point in time in between determined by interpolation. Some applicants also indicated that they plan to replace the hold-down spring with Type 403 stainless steel material that is not susceptible to stress relaxation, thus does not require physical measurements.

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

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

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

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

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

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

A/LAI 6 stated:

> MRP-227 does not propose to inspect the following inaccessible components: the B&W core barrel cylinders (including vertical and circumferential seam welds), B&W former plates, B&W external baffle-to-baffle bolts and their locking devices, B&W core barrel-to-former bolts and their locking devices, and B&W core barrel assembly internal baffle-to-baffle bolts. The MRP also identified that although the B&W core barrel assembly internal baffle-to-baffle bolts are accessible, the bolts are non-inspectable using currently available examination techniques.
>
> Applicants/licensees shall justify the acceptability of these components for continued operation through the period of extended operation by performing an evaluation, or by proposing a scheduled replacement of the components. As part of their application to implement the approved version of MRP-227, applicants/licensees shall provide their justification for the continued operability of each of the inaccessible components and, if

necessary, provide their plan for the replacement of the components for NRC review and approval.

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

a) The core barrel cylinder and welds

b) The former plates; and

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

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

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

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

A/LAI 7 stated:

> The applicants/licensees of B&W, CE, and Westinghouse reactors are required to develop plant-specific analyses to be applied for their facilities to demonstrate that B&W IMI guide tube assembly spiders and CRGT spacer castings, CE lower support columns, and Westinghouse lower support column bodies will maintain their functionality during the period of extended operation or for additional RVI components that may be fabricated from CASS, martensitic stainless steel or precipitation hardened stainless steel materials. These analyses shall also consider the possible loss of fracture toughness in these components due to thermal and irradiation embrittlement, and may also need to consider limitations on accessibility for inspection and the resolution/sensitivity of the inspection techniques. The requirement may not apply to components that were previously evaluated as not requiring aging management during development of MRP-227. That is, the requirement would apply to components fabricated from susceptible materials for which an individual licensee has determined aging management is required, for example during their review performed in accordance with Applicant/Licensee Action Item 2. The plant-specific analysis shall be consistent with the plant's licensing basis and the need to maintain the functionality of the

components being evaluated under all licensing basis conditions of operation. The applicant/licensee shall include the plant-specific analysis as part of their submittal to apply the approved version of MRP-227.

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

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

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

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

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

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

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

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

c) The discussion of LSC buckling in the redundancy analysis adequately addressed buckling of an LSC subject only to a compressive axial load but did not address the effect of eccentric[6] loading in LSC buckling; and the discussion of the LSS assembly dynamic response described in the report adequately shows there is little change in the dynamic response of the LSS assembly due to failed LSCs.

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

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

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

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

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

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

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

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

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

are not “no additional measures components,” loss of fracture toughness has been adequately addressed.

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

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

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

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

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

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

A/LAI 8 states that:

> Applicants/licensees shall make a submittal for NRC review and approval to credit their implementation of MRP-227, as amended by this SE, as an AMP for the RVI components at their facility. This submittal shall include the information identified in Section 3.5.1 of [the staff’s final SE of MRP-227, Revision 0]. The items detailed in Section 3.5.1 are 1) an RVI AMP, 2) RVI inspection plan, 3) FSAR supplement, 4) any necessary TS changes needed to manage aging of RVI, and 5) any TLAAs related to

RVI. For licensees that already have a renewed license, A/LAI 8 only requires the first two items. The fifth item also states in part that for those CUF analyses that are TLAAs, the applicant may use the PWR Vessel Internals Program as the basis for accepting these CUF analyses in accordance with 10 CFR 54.21(c)(1)(iii) only if the RVI components in the CUF analyses are periodically inspected for fatigue-induced cracking in the components during the period of extended operation, and that the periodicity of the inspections of these components shall be justified to be adequate to resolve the TLAA. The fifth item also included wording requiring that the fatigue TLAAs for RVI address the effects of the reactor water environment.

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

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

The additional information in Item 5 regarding the periodicity of inspections for fatigue would be addressed through the NRC staff review of the disposition of the fatigue TLAA for RVI. The statement in Item 5 that "To satisfy the evaluation requirements of ASME Code, Section III, Subsection NG-2160 and NG-3121, the existing fatigue CUF analyses shall include the effects of the reactor coolant system water environment" adds an additional technical requirement to the review of the fatigue TLAA for RVI.

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

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

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

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

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

4.0 APPLICANT/LICENSEE ACTION ITEMS

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

A/LAI 1

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

5.0 CONCLUSIONS

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

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

6.0 REFERENCES

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48. Enclosure: WAAP-8828-P, Revision 0, "Lower Radial Support System (LRSS) Clevis Inserts and Attachment Bolts Design and Safety Function" (Proprietary), March 26, 2014 (ADAMS Accession No. ML14170A264)

49. Indian Point Nuclear Generating Unit Nos. 2 & 3 - Reply to Request for Additional Information Regarding the License Renewal Application, September 27, 2013 (ADAMS Accession No. ML13277A007)

50. Indian Point, Units 2 & 3 - Reply to Request for Additional Information Regarding the License Renewal Application, June 9, 2014 (ADAMS Accession No. ML14176A159)

51. Final Safety Evaluation of Action Items 1 And 7 From Topical Report MRP-227-A, "Materials Reliability Program: Pressurized Water Reactor Internals Inspection And Evaluations Guideline," January 29, 2018 (ADAMS Accession No. ML18016A008)

52. Oconee Nuclear Station, Units 1, 2, and 3, Issuance of Amendments Regarding Inspection Plan for Reactor Vessel Internals (TAC Nos. ME9024, ME9025, and ME9026), June 9, 2015 (ADAMS Accession No. ML15050A671)

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

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

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

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

57. U. S. Nuclear Regulatory Commission Letter, “Summary of January 22-23, 2013, Closed Meeting with the Electric Power Research Institute and Westinghouse,” February 21, 2013 (ADAMS Accession No. ML13042A048/ML13043A062)

58. U. S. Nuclear Regulatory Commission Letter, “Summary of February 25, 2013, Telecom with the Electric Power Research Institute and Westinghouse Electric Company,” March 15, 2013 (ADAMS Accession No. ML13067A262)

59. U. S. Nuclear Regulatory Commission Letter, “Summary of May 21, 2013, Public Meeting Regarding Pressurized Water Reactor (PWR) Vessel Internals Inspections,” June 24, 2013 (ADAMS Accession No. ML13164A126)

60. U. S. Nuclear Regulatory Commission Presentation: “Status of MRP-227-A Action Items 1 and 7,” June 5, 2013 (ADAMS Accession No. ML13154A152).

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

62. Electric Power Research Institute (EPRI), MRP-227-A Applicability Template Guideline, October 14, 2013 (ADAMS Accession No. ML13322A454) (MRP Letter 2013-025)

63. Evaluation of WCAP-17780-P, "Reactor Internals Aging Management MRP-227-A Applicability for Combustion Engineering and Westinghouse Pressurized Water Reactor Designs," and MRP-227-A, Applicability Guidelines for CE, November 7, 2014 (ADAMS Accession No. ML14309A484)

64. PWROG-15105-NP "PA-MSC-1288 PWR RV Internals Cold-Work Assessment, Materials Committee," April 30, 2016 (ADAMS Accession No. ML17075A195)

65. PWR Owners Group Submittal of PWROG-15105-NP, Revision 0, "PWR RV Internals Cold-Work Assessment" to the NRC for Information, June 15, 2016 (ADAMS Accession No. ML16222A299)

66. Staff Assessment of PWROG-15105, April 21, 2017 (ADAMS Accession No. ML17081A010)

67. 1013234, "Materials Reliability Program: Screening, Categorization, and Ranking of Reactor Internals Components for Westinghouse and Combustion Engineering PWR Design (MRP-191)," November 30, 2006 (ADAMS Accession No. ML091910130)

68. PWR Owners Group - Submittal of PWROG-14048-P, Revision 0, "Functionality Analysis: Lower Support Columns" to the NRC for Information Only (PA-MSC-1103), March 13, 2015 (ADAMS Accession No. ML15077A113)

69. Pressurized Water Reactor Owners Group Report No. PWROG-14048-P, “Functionality Analysis: Lower Support Columns,” Revision 0 (ADAMS Accession No. ML15077A114)

70. Summary Assessment Of PWROG-14048-P, Revision 0, "Functionality Analysis: Lower Support Columns" December 17, 2015 (ADAMS Accession No. ML15334A462)

71. PWR Owners Group - Submittal of PWROG-14048-P, Revision I, "Functionality Analysis: Lower Support Columns" to the NRC for Information, March 1, 2017 (ADAMS Accession No. ML17066A266)

72. Staff Assessment of Report PWROG-14048-P, Revision 1, "Functionality Analysis: Lower Support Columns," August 30, 2017 (ADAMS Accession No. ML17242A003)

73. Staff Assessment of the Pressurized Water Reactor Owner's Group Report PWROG-15032-NP, Revision 0, "PA-MSC-1288 Statistical Assessment of PWR RIV Internals CASS Materials" (TAC No. MF7223), August 25, 2016 (ADAMS Accession No. ML16222A254)

74. Final BWRVIP-234 Safety Evaluation and Transmittal Letter, June 22, 2016 (ADAMS Accession No. ML16096A002)

75. Three Mile Island Nuclear Station, Unit 1 - Non-Proprietary, Staff Assessment of Action Item 7 Regarding Inspection Plan for Reactor Internals (EPID L-2016-LLL-0002), July 12, 2018 (ADAMS Accession No. ML18159A108)

76. Models of Irradiation-Assisted Stress Corrosion Cracking of Austenitic Stainless Steels in Light Water Reactor Environments; Volume 1: Disposition Curves Developments, EPRI, Palo Alto, CA: 2014. 3002003103

77. PWROG-17071-NP, "WCAP-17096-NP-A Interim Guidance," March 31, 2018 (ADAMS Accession No. ML18204A166)

78. PWR Owners Group Submittal of PWROG-17071-NP, "WCAP-17096-NP-A Interim Guidance" to the NRC for Information Only (PA-MSC-1567), July 12, 2018 (ADAMS Accession No. ML18204A165)

79. May 2018 Materials Programs Technical Information Exchange Public Meeting (ADAMS Package Accession No. ML18144A252)

80. CRDM Thermal Sleeve LIC-504 - Rev 2, September 27, 2018 (ADAMS Accession No. ML18249A107)

81. NRC Information Notice 2018-10: "Thermal Sleeve Flange Wear Leads to Stuck Control Rod at Foreign Nuclear Plant," August 29, 2018 (ADAMS Accession No. ML18214A710)

82. EPRI Report 1015013, "Materials Reliability Program: PWR Internals Age-Related Material Properties, Degradation Mechanisms, Models, and Basis Data - State of Knowledge (MRP-211)," December 31, 2007 (ADAMS Accession No. ML093020614)

83. MRP 2014-009, Biennial Report of Recent MRP-227-A Reactor Internals Inspection Results (Project 694), May 12, 2014 (ADAMS Accession Nos. ML14135A383, ML14135A384, and ML14135A385)

84. MRP 2016-008, Biennial Report of MRP-227-A Reactor Internals Inspection Results, May 18, 2016 (ADAMS Accession No. ML16144A789)

85. EPRI - 2018 Biennial Report of Recent MRP-227-A Reactor Internals Inspection Results, July 19, 2018 (ADAMS Accession No. ML18204A161)

Attachment: Comment Resolution Table

Principal Contributor: Jeff Poehler, NRR

Date:

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

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

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