UNITED STATES
NUCLEAR REGULATORY COMMISSION
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May 15, 2017

Brian Burgos
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SUBJECT: REQUEST FOR ADDITIONAL INFORMATION FOR ELECTRIC POWER RESEARCH INSTITUTE TOPICAL REPORT MRP-227, REVISION 1, "MATERIALS RELIABILITY PROGRAM: PRESSURIZED WATER REACTOR INTERNALS INSPECTION AND EVALUATIONS GUIDELINE" (CAC NO. MF7740)

Dear 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." Upon review of the information provided, the NRC staff has determined that additional information is needed to complete the review. The request for additional information (RAI) questions are provided in the enclosure to this letter.

In an email exchange between Mr. Kyle Amberge representing EPRI and me, we agreed that the NRC staff will receive your response to the enclosed RAI questions by September 27, 2017.

If you have any questions regarding the enclosed RAI questions, please contact me at 301-415-7297 or Joseph.Holonich@nrc.gov.

Sincerely,

/RA/

Joseph J. Holonich, Senior Project Manager
Licensing Processes Branch
Division of Policy and Rulemaking
Office of Nuclear Reactor Regulation

Project No. 669

Enclosure:
RAI questions

SUBJECT: REQUEST FOR ADDITIONAL INFORMATION FOR ELECTRIC POWER RESEARCH INSTITUTE TOPICAL REPORT MRP-227, REVISION 1, “MATERIALS RELIABILITY PROGRAM: PRESSURIZED WATER REACTOR INTERNALS INSPECTION AND EVALUATIONS GUIDELINE” (CAC NO. MF7740) DATED: MAY 15, 2017

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REQUEST FOR ADDITIONAL INFORMATION

FROM THE OFFICE OF NUCLEAR REACTOR REGULATION

MATERIALS RELIABILITY PROGRAM: PRESSURIZED WATER REACTOR INTERNALS

INSPECTION AND EVALUATION GUIDELINE (MRP-227 REVISION 1)

ELECTRIC POWER RESEARCH INSTITUTE PROJECT NUMBER 669

RAI 1

In MRP-227, Rev. 1 (Ref. 1), for the following Babcock & Wilcox (B&W) primary components, the schedule for the initial (baseline) examination changed from "during the next 10-year ISI [inservice inspection]" to "during the next 10-year ISI interval."

- B2. Control Rod Guide Tube (CRGT) Assembly - spacer castings
- B3. Vent Valve Assembly
    - A. Vent valve top retaining ring
    - B. Vent valve bottom retaining ring
- B10. Core Barrel Assembly-Baffle plates
- B11. Core Barrel Assembly - Locking devices, including locking welds, of baffle-to-Former bolts and internal baffle-to-baffle-bolts

Clarify what this means; for example, does "during the next 10-year ISI" mean during the next scheduled 10-year ISI examination of the reactor vessel internals (RVI), or does it mean sometime during the next 10-year ISI interval? If the latter, does that mean these examinations may not be performed until up to 20 years from now, if the current 10-year ISI interval started today? If this is the case, justify waiting up to 20 years to perform the baseline examination.

RAI 2

For MRP-227-A item in Table 4-1, "B&W Plants Primary Components:"

Plenum Cover Assembly & Core Support Shield Assembly

Plenum cover weldment rib pads
Plenum cover support flange
Core Support Shield (CSS) top flange

The revised item description in MRP-227, Rev. 1, Table 4-1 is:

"B1.Plenum Cover Assembly & Core Support Shield Assembly"
a. Plenum cover weldment rib pads
b. Plenum cover support flange
c. Plenum cover support ring
d. CSS top flange"

The examination coverage changed from "Determination of differential height of top of plenum rib pads to reactor vessel seating surface, with plenum in reactor vessel" to "Determination of differential height of top of plenum rib pads/plenum cover support ring location to reactor vessel seating surface, with plenum in reactor vessel."

The change to this item was to add the plenum cover support ring as a subcomponent and to add this subcomponent as an additional reference point for the physical measurement.

The plenum cover support ring appears to be a new subcomponent added in MRP-227, Rev. 1. The plenum cover support ring is addressed in MRP-189, Rev. 1, "Screening, Categorization, and Ranking of B&W-Designed PWR Internals Component Items" (Ref. 6), and was determined to be Category A. The plenum cover assembly – weldment rib pads and plenum cover assembly – support flange were determined to be Category C for wear in MRP-189. Therefore, the NRC staff requests that the Electric Power Research Institute (EPRI) clarify why the plenum cover support ring was added as a subcomponent and how and why the support ring was added as a reference location for making the physical measurements.

RAI 3

In Table 4-1, the "Control Rod Guide Tube Assembly – CRGT," spacer castings previously had no expansion link. An expansion link to the vent valve bodies has now been added in MRP-227, Rev. 1. The vent valve bodies were not an expansion component in MRP-227-A. According to MRP-189, Rev. 1, the vent valve bodies are cast austenitic stainless steel (CASS), as are the CRGT spacer castings. Since the vent valve bodies were previously a no additional measures component, the U.S. Nuclear Regulatory Commission (NRC) staff requests that EPRI explain why the vent valve bodies were made an expansion component for the CRGT spacer castings.

RAI 4

In Table 4-1, the schedule for the initial (baseline) ultrasonic (UT) examination of the Core Barrel Assembly - Baffle-to-former Bolts changed from "no later than two refueling outages from the beginning of the license renewal period" to "volumetric (UT) examination during the next 10-year ISI interval." Since it is not clear when the next ten-year ISI interval starts (it could be up to ten years from the current date), this could result in the baseline examination being significantly later than MRP-227-A would require. It was not clear to the NRC staff whether this assumes all six operating B&W units have completed baseline UT examinations already. The NRC staff, therefore, requests EPRI provide the following information:

1. Does the initial baseline UT examination schedule for the baffle-to-former bolts in MRP-227, Rev. 1 assume an examination of baffle-to-former bolts has been completed within two refueling outages from the beginning of the period of extended operation?

2. If not, justify the change in the schedule for the initial (baseline) UT examination of the baffle-to-former bolts.

RAI 5

The required examination coverage in Table 4-2, "CE Plants Primary Components," and Table 4-3, "Westinghouse Primary Components," for four weld items, all of which are classified as high-consequence components in MRP-191, "Materials Reliability Program: Screening, Categorization, and Ranking of Reactor Internals Components for Westinghouse and

Combustion Engineering PWR Design (MRP-191)," has been changed from 100 percent of the accessible surfaces of the [weld]" in MRP-227-A, to essentially a 25 percent sample of the weld circumference in MRP-227, Rev. 1. Table 1 below lists the old and new component item designations and the revised coverage requirement in MRP-227, Rev. 1.

Table 1 – Combustion Engineering and Westinghouse Core Support Barrel/Core Barrel Welds with Coverage Reduction in MRP-227, Rev. 1

MRP-227-A Item	Equivalent MRP-227, Rev. 1 Item	MRP-227, Rev. 1 Coverage Requirement
Core Support Barrel Assembly – Upper (core support Barrel) flange weld	C.5 Core Support Barrel Assembly Upper Flange Weld (UFW)	A minimum of 25% of the circumference of the UFW and adjacent base metal shall be examined
Core Support Barrel Assembly – Lower Cylinder Girth Welds	C6. Core Support Barrel Assembly – Middle Girth Weld (MGW)	A minimum of 25% of the OD circumference of the MGW and adjacent base metal shall be examined
Core Barrel Assembly – Upper core barrel flange weld	W3.Core Barrel Assembly – Upper flange weld (UFW)	A minimum of 25% of one side of the circumference of the surface of the UFW and adjacent base metal shall be examined
Core Support Barrel Assembly – Upper and Lower Core Barrel Cylinder Girth Welds	W4. Core Barrel Assembly – Lower Girth Weld (LGW)	A minimum of 25% of the OD circumference of the LGW and adjacent base metal shall be examined

For the original items in MRP-227-A, Note 4 clarified that a minimum of 75 percent of the total weld length (examined + unexamined) including coverage consistent with the Expansion criteria in [Table 5-2 or table 5-3], must be examined from either the inner or outer diameter for inspection credit. In MRP-227, Rev. 1, Note 5 to Table 4-2 and Note 8 to Table 4-3 state that "Examination coverage requires 25% of the circumference of either the inside diameter or the outside diameter of the weld." Note 6 to Table 4-2 and Note 10 to Table 4-3 state that "The stated coverage requirement is the minimum if no significant indications are found. However, the Examination Acceptance criteria in Section 5 require that additional coverage must be achieved in the same outage if significant flaws are found. This contingency should be considered for inspection planning purposes."

MRP-227, Rev. 1 contains a discussion of the inspection strategy for Westinghouse/Combustion Engineering (CE) core barrel weld sampling on p. 4-10 through 4-11 that pertains to the welds listed above. The discussion focuses on two elements: (1) A discussion of the probability of detecting an active cracking mechanism if a 25 percent sample of the weld is examined, on both a single plant and fleet-wide basis; (2) Focusing the 25 percent sample on the accessible portion of the weld most likely to exhibit cracking.

The NRC staff has several concerns related to the reduction of the required examination coverage for the welds listed in Table 1:

- The NRC staff is concerned that the reduced examination coverage is insufficient to provide reasonable assurance of component functionality considering that these welds are high consequence of failure items, which are not part of a redundant population.

- The discussion on pages 4-10 to 4-11 of MRP-227, Rev. 1 appears to describe some elements of a technical basis, but more detail is needed by the NRC staff to determine the adequacy of the technical basis.

- A determination of the most likely accessible portion of the weld to experience cracking is not required in Table 4-2 or Table 4-3. The discussion on pages 4-10 to 4-11 is not part of the report designated as NEI 03-08 “needed” guidance. Therefore, there is no guarantee licensees would perform such a determination.

- Even if a determination of the most likely accessible portion of the weld to experience cracking, is made, there may be significant uncertainty associated with such a determination and cracking may still be more likely to initiate in inaccessible portions, such as the weld ID.

- Coverage appears to be inconsistent between the UFW for the CE versus Westinghouse designs, with CE apparently required to examine both sides and Westinghouse only one side.

The NRC staff, therefore, requests the following information:

- Provide a technical justification for the reduction in the required examination coverage from 100 percent (minimum 75%) to 25 percent, for the component items listed in Table 1. If the technical justification relies in whole or part upon a statistical analysis, provide the detailed statistical analysis. The technical justification for the reduction in examination coverage should provide reasonable assurance that (1) the functionality of the core barrel will be maintained and (2) the structural integrity of the core barrel will be maintained to ensure safe shutdown of the reactor during the period of extended operation (PEO).

- Clarify whether the justification for reduction in the required examination coverage relies on the assumption that licensees will perform a plant-specific determination of the most likely portion of the weld to experience cracking.

- Discuss how it can be assured that the 25 percent sample of each weld examined will be selected based on an evaluation of the most likely accessible portion of the weld to exhibit cracking, since Table 4-2 and 4-3 do not require such an evaluation.

- Discuss how the proposed 25 percent sample examination coverage accounts for the possibility of cracking initiating on the opposite side of the weld from the side examined or in a portion of the component that is inaccessible.

- For C5., “Core Support barrel Assembly Upper Flange Weld (UFW),” clarify whether 25 percent of bolt sides of the weld are to be examined. If both sides are to be examined, explain the inconsistency with W3. Core Barrel Assembly UFW, for which MRP-227, Rev. 1 only requires one side to be examined.

<u>RAI 6</u>

In Table 4-1, Item B11., “Core Barrel Assembly – Locking Devices,” including locking welds, of baffle-to-former bolts and internal baffle-to-baffle bolts, has applicable aging mechanisms of irradiation assisted stress corrosion cracking (SCC) (IASCC), irradiation embrittlement (IE) including the detection of missing, non-functional, or removed locking devices or welds, and has as an Expansion link "locking devices, including locking welds, of the external baffle-to-baffle bolts and core barrel-to-former bolts." However, in MRP-227, Rev. 1, a new Note 8 has been added for the expansion link, which states that "the aging degradation mechanism of IASCC is only applicable to the baffle-to-former bolt and internal baffle-to-baffle bolt locking devices, not the baffle-to-former bolt and internal baffle-to-baffle bolt locking device welds." However, under the expansion link column in Table 4-1, the expansion link for Item B11 is described as locking devices, including locking welds, of the external baffle-to-baffle bolts and core barrel-to-former bolts.

The NRC staff therefore requests the following information:

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

RAI 7

For all the welds listed in Table 1 except for Item W4., "Core Barrel Assembly – Lower Girth Weld (LGW)," the examination acceptance and expansion criteria in Table 5-2 and Table 5-3 require the inspection coverage to be extended to include 100% of the accessible length of the weld during the same refueling outage, if there is confirmed detection of a surface breaking linear indication in that weld. Should this expansion also be applied to Item W4? If not, provide a technical justification.

RAI 8

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

Westinghouse Nuclear Safety Advisory Letter NSAL-16-1, Revision 1 "Baffle-Former Bolts (Ref. 2)," categorized all Westinghouse and CE design RVI with respect to susceptibility to BFB degradation. EPRI interim guidance in MRP Letter 2016-022 (Ref. 3) endorsed the recommendation of Westinghouse NSAL 16-1 that 4-loop, downflow plants with Type 347 bolts complete baseline UT examinations of BFB by the next refueling outage. These baseline examinations are expected to be complete by the end of 2017. EPRI interim guidance in MRP Letter 2017-002 (Ref. 4) endorsed the NSAL 16-1 guidance for 2-loop and 3-loop downflow plants (Tier 2), which calls for a review of previous UT examination results from these plants for evidence of clustering, with UT examination the next refueling outage if evidence of clustering is seen. The EPRI interim guidance does not provide any guidance on how the subsequent examination interval is to be determined for Tier 1 and Tier 2 plants. The default subsequent examination interval in MRP-227, Revision 1, remains ten years. However, the NRC staff is concerned that a default subsequent examination interval of ten years may not be appropriate for the highest susceptibility groups of plants.

If BFB degradation is found, an engineering evaluation is required. MRP-227, Revision 1, Section 7.5 defines as NEI 03-08 "needed" guidance that, if examination results that do not meet the examination acceptance criteria defined in Section 5 of these guidelines shall be recorded and entered in the owner's plant corrective action program and dispositioned, and that such engineering evaluations shall be conducted in accordance with NRC-approved evaluation methods (i.e. ASME Code Section XI, WCAP-17096-NP or equivalent method). Current NRC-approved guidance for determining the subsequent examination interval for BFBs is found in WCAP-17096-NP-A, Rev. 2 "Reactor Internals Acceptance Criteria Methodology and Data Requirements" (Ref. 5), pages E-42 to E-43, which allows a subsequent examination interval of 10 years provided that no more of 50 percent of the initial margin with respect to the minimum required number of bolts is found degraded at the initial UT examination. However, WCAP-17096-NP-A does not provide any guidance for determining the subsequent examination interval if greater than 50 percent of the bolts constituting the margin are degraded, even if degraded bolts are replaced. In addition, the guidance in WCAP-17096-NP-A for determining the subsequent examination interval does not take into account the possibility of clustering of degraded bolts as was seen in the 4-loop plants in 2016, and did not account for the large extent of BFB degradation seen in certain plants.

The NRC staff requests that EPRI:

a. Discuss whether revised guidance for BFB needs to be incorporated into MRP-227, Rev. 1. If not, why not?
b. If such guidance should be incorporated, provide specifics on the initial examination coverage and schedule, and on how the subsequent examination coverage and timing would be determined.
c. Considering the recent OE with BFB degradation, justify that a ten-year subsequent examination interval remains appropriate for BFB. This justification should consider the possible effects of clustering.
d. How will the schedule for subsequent examination be determined if examination results show that greater than 50 percent of the numerical margin of bolts is degraded?
e. Provide a justification that the criteria allowing subsequent examination of BFB may be performed in ten years, provided 50 percent or less of the numerical margin of BFB is degraded, is still appropriate considering the discovery of clustering of degraded BFB, and the discovery of more extensive BFB degradation than expected.

<u>RAI 9</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% 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. There are differences in required examination coverage for the core support column components for the two plant design variations. In addition, the component in Westinghouse-design RVI with the same function is an Expansion component whereas the CE core support columns are a Primary component.

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

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

To resolve these discrepancies, the NRC staff requests the following information:

a. Justify the required coverage of 25 percent as visible from above the core plate for Item C8 and W4.4 is sufficient to provide reasonable assurance of functionality.
b. Justify the use of VT-3 examination instead of EVT-1 to detect cracking.
c. Clarify the meaning of “25% of column assemblies as visible using a VT-3 examination from above the lower core plate.” Does this mean that 1) only 25 percent of the total number of columns visible need to be inspected, 2) 25 percent of the total number of columns (visible and not visible) must be examined to claim credit for the examination, or that 3) 25 percent of the total columns should be inspected if this number is visible? Should all columns visible from above the core plate be examined, or just enough to constitute 25 percent of the total population (visible plus not visible).
d. What expansion of the examination scope to the remaining columns will be conducted if degradation is observed in the 25 percent sample?
e. For CE-design RVI, explain why examination of the core support columns is specified only for plants with full-height bolted shroud plates and not for plants with core shrouds assembled in two vertical sections.
f. Explain why the core support columns are a Primary component for CE plants but the component in Westinghouse plants with the same function (lower core support columns) is an Expansion component.

<u>RAI 10</u>

In Table 4-2, for Item C12., “Lower Support Structure – Deep Beams,” and Table 4-5, Item C5.4., “Lower Support Structure - Lower Core support Beams,” the examination coverage has been changed to 25 percent of the total number of beam-to-beam welds. The examination coverage in MRP-227-A for the Lower Support Structure - Deep Beams does not specify a percentage of beam-to-beam welds that must be examined, but it is implied that 100 percent of the welds should be examined. The examination coverage in MRP-227-A for the lower core support beams is 100 percent of accessible surfaces. Because both components are high consequence of failure components, the NRC staff is concerned that the reduced examination coverage is insufficient to ensure functionality of the components.

The NRC staff therefore requests the following information:

a. Provide a justification for the reduction in coverage for these two items. The technical justification for the reduction in examination coverage should provide reasonable assurance that (1) the functionality of the components will be maintained and (2) the structural integrity of the components will be maintained to ensure safe shutdown of the reactor during the PEO.

b. What expansion to the remaining beam-to-beam welds will be conducted if degradation is found in the initial 25 percent inspection sample?

RAI 11

In Table 4-4, B&W Plants Expansion Components, Item B7.1 Core Barrel Assembly - Upper Thermal Shield (UTS) bolts and their locking devices and Item B7.2 Core Barrel Assembly - Surveillance specimen holder tube (SSHT) bolts and their locking devices, had changes to the “Effect (mechanism)” information. Specifically, irradiation creep/irradiation stress relaxation (IC/ISR)/Wear/Fatigue were added for the SSHT bolts.

Note 7 has also been added to Table 4-4 indicating that this table entry for the SSHT bolts also includes the aging degradation mechanisms of ISR/IC/wear/fatigue for the compression collar and washer for the SSHT bolt. The compression collars for the SSHT bolt are not included in the screening and failure mode, effects and criticality analysis (FMECA) documented in MRP-189, "Materials Reliability Program: Screening, Categorization, and Ranking of B&W-Designed PWR Internals Component Items” (Ref. 6) and MRP-190, "Materials Reliability Program: Failure Modes, Effects, and Criticality Analysis of B&W-Designed PWR Internals” (Ref. 7).

The NRC staff, therefore, requests EPRI:

a. Explain why the new aging mechanisms of IC/ISR/Wear/Fatigue was added for the SSHT bolts.

b. Clarify whether the compression collars were left out of the screening and FMECA process as an oversight, or whether the compression collars are the same as the SSHT bolt locking cups and tie plates that are included in the screening and FMECA. If the latter, explain why the screening and FMECA results for these components changed.

RAI 12

In Table 4-5, for plant designs with core shrouds assembled with full-height shroud plates, the core shroud assembly, remaining axial welds, ribs and rings has been split into two items: C3.1, “Remaining axial welds,” and C3.2, “Ribs and rings.” The coverage for these two items is different, 75 percent for the remaining axial weld length and adjacent base metal as visible from the core side of the shroud other than that already inspected under the primary link, and 25 percent of the Ribs and rings. Also, in MRP-227, Rev. 1, Core Shroud Assembly (Welded) Item C2.1, “Remaining Axial Welds,” is a new expansion component applicable to plant designs with core shrouds assembled in two vertical sections. The coverage for Item C2.1 is the same as for Item C3.1. In MRP-227-A, the coverage for the axial welds, ribs and rings was “axial welds seams” other than the core shroud reentrant corner welds at the core mid-plane, plus ribs and rings. Although the extent of coverage required has been quantified, no justification is provided for the examination coverages for the remaining axial welds, or the ribs and rings. Also, in Figure 4-37, it is not clear if the core shroud assembly can be removed from the core support barrel assembly to allow examination of the ribs and rings.

a. For Item C2.1 and 3.1, does 75 percent of the remaining axial weld length for the remaining axial welds mean a minimum of 75 percent of the total accessible plus inaccessible length of these welds must be examined to claim examination credit?

b. Justify the 25 percent sample size for the ribs and rings (Item C3.2).

c. Clarify whether the ribs and rings are accessible for visual examination.

RAI 13

The following four areas pertaining to Tables 4-1 through 4-3 could be informed by OE related to the actual accessibility achieved for primary system components during baseline inspections:

1. In MRP-227, Rev. 1, has OE been used to modify or clarify examination coverage requirements of MRP-227-A based on the actual accessibility achieved during the examinations completed to date? If so, identify the components that have had examination coverage revised based on OE, and describe the reason for the change. If coverage requirements have not been revised based on OE, justify why this has not been done.
2. Has OE with actual coverage achieved resulted in any primary component that was previously considered to be accessible being reclassified as inaccessible, either because of the percentage of the component surface area, length, or population that is accessible was insufficient to provide reasonable assurance of functionality, or because insufficient coverage was achieved of the most likely portion of the component to exhibit degradation? Identify any primary components that have been reclassified as inaccessible and identify what alternate measures, such as an engineering analysis, were taken to provide reasonable assurance of component functionality.
3. For primary components reclassified as inaccessible, were the expansion links reevaluated for these components?
4. For any primary components reclassified as inaccessible, were alternate primary components selected?

RAI 14

The inspection method and coverage for two Westinghouse expansion components, the Upper Internals Assembly – Item W4.1., "Upper Core Plate and the Lower Internals Assembly," – Item W3.4., "lower support forging or casting," has been changed from EVT-1 examination of 100 percent of accessible surfaces to VT-3 examination of 25% of the bottom (non-core side) surfaces. However, both of these items are non-redundant components. The NRC staff does not generally consider VT-3 examination to be an adequate examination method for non-redundant components unless these components are highly flaw tolerant. In addition, the examination coverage has been reduced. The NRC staff is concerned that the reduced examination coverage is not sufficient to provide reasonable assurance of component functionality, considering that these are high consequence of failure components.

Therefore, the NRC staff requests that EPRI:

a. Justify the use of VT-3 examination for these components;

b. Justify the reduction in examination coverage from 100 percent to 25 percent. The technical justification for the reduction in examination coverage should provide reasonable assurance that (1) the functionality of the components will be maintained and (2) the structural integrity of the components will be maintained to ensure safe shutdown of the reactor during the PEO.

c. Is it intended that if the examination of the 25 percent sample of these items reveals indications, the examination coverage will be expanded to include the remaining accessible surfaces of these components? If not, why not?

RAI 15

Applicant/licensee Action Item (A/LAI) 3 was included in the NRC staff's final safety evaluation of MRP-227, Rev. 0 (Ref. 8) because MRP-227-A did not provide adequate guidance for applicants/licensees to document the specifics of the plant-specific existing programs in plant-specific RVI programs. With respect to Westinghouse existing plant-specific programs, MRP-227-A stated that "The guidance for guide tube support pins (split pins) is limited to plant-specific recommendations and thus have no generic reference. Subsequent performance monitoring should follow the supplier recommendations. Thus, they are not included in Table 4-9. The owner should review their specific design, upgrade status, and asset management plans for guide tube support pins (split pins)."

For Westinghouse split pins, similar guidance is included in MRP-227, Rev. 1 to that in MRP-227-A. The revised guidance states, "Additionally, in Westinghouse –design plants, the originally installed alloy X750 guide tube support pins (split pins) have been typically replaced with components with improved designs and less susceptible materials. The plant owner should review their specific design, upgrade status, and asset management plans for guide tube support pins (split pins). Thus the guide tube support pins (split pins) are not included in Table 4-9." However, the guidance is not sufficient because it does not specify that an applicant or licensee must include the specifics of the aging management program for split pins in its plant-specific RVI program. Also, the revised wording appears to imply that aging management is only necessary for Alloy X-750 split pins. Further, in Table 3-3, the material for the guide tube support pins is listed as Alloy X-750, implying that Type 316 split pins are a no additional measures component.

Therefore, the NRC staff requests that EPRI:

a) Clarify if type 316 stainless steel split pins require a plant-specific aging management program, or whether they are a "no additional measures component." Modify the wording of section 4.4 of MRP-227, Rev. 1 as necessary.

b) Discuss whether it would be appropriate to include a requirement in MRP-227, Rev. 1 that the specific aging management program for split pins be documented in the plant-specific RVI program, including the replacement and/or inspection schedule, replacement material, examination method and coverage, technical basis for the replacement schedule or the remaining life of the split pins (if already replaced), and technical basis for the inspection schedule or lack of inspections.

RAI 16

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

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

Therefore, the NRC staff requests that EPRI:

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

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

RAI 17

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

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

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

Does the change from “or” to “and” mean degradation now has to be exhibited in both the locking devices for the baffle-to-former bolts and the locking devices for the internal baffle-to-baffle bolts for the expansion to be required, whereas in MRP-227-A the expansion would be required if only one of these items exhibited degradation? If so, justify the changes.

RAI 18

In Table 4-4, the Lower Grid Assembly – Item B10.3., “Lower Grid Rib Section,” has been added as an additional Expansion link for Primary Item B10., “Core Barrel Assembly – Baffle Plates. Lower Grid Assembly,” – Item B10.3., “Lower Grid Rib Section,” was not included in MRP-227-A as either a Primary or Expansion component. The NRC staff therefore requests EPRI explain why this item has apparently been recategorized from “no additional measures” to “expansion.”

RAI 19

In Reference 9, Westinghouse submitted a notification pursuant to Title 10 of the *Code of Federal Regulations* (10 CFR) Part 21, that notified the NRC of a potential significant safety hazard due to guide card wear in four Westinghouse units that use ion nitride rod cluster control assemblies (RCCAs) in conjunction with 17x17 A or 17 x 17 AS style guide tubes. Guide card wear in these plants may occur more rapidly than predicted by WCAP-17451-P, Revision 1, "Reactor Internals Guide Tube Wear - Westinghouse Domestic Fleet Operational Projects," (Ref. 10) which is referenced in MRP-227, Rev. 1 with respect to the examination schedule, method, and coverage for CRGT guide plates (guide cards) in Westinghouse-design RVI. The

NRC staff was also informed of a Westinghouse Nuclear Safety Advisory Letter (NSAL), issued in January 2017 (currently non-public), which addressed the accelerated wear issue and defined an accelerated schedule for the baseline guide card wear measurements at the four affected units.

The NRC staff requests that EPRI discuss how MRP-227, Rev. 1 and/or WCAP-17451-P, Rev. 1 should be modified to address the OE discussed in the 10 CFR Part 21 notification related to guide cards (Ref. 9).

RAI 20

For a number of Primary and Expansion weld items in Tables 4-2, 4-3, 4-5, and 4-6, the revised examination coverage in MRP-227, Rev. 1 specifies a percentage of the weld length or circumference "and adjacent base metal" shall be examined. The weld items are listed in Table 2 below. The NRC staff requests EPRI define what extent of the adjacent base metal must be examined (e.g., a certain distance from the weld fusion line or centerline).

Table 2 – Weld Items with Adjacent Base Metal to be Examined

Table	Item
4.2.	C5. Core Support Barrel Assembly Upper Flange Weld (UFW)
4.2	C6. Core Support Barrel Assembly – Middle Girth Weld (MGW)
4.3	W3. Core Barrel Assembly – Upper flange weld (UFW)
4.3	W4. Core Barrel Assembly – Lower Girth Weld (LGW)
4.5	C5.1 Core Support Barrel Assembly – Lower Girth Weld (LGW)
4.5	Core Support Barrel Assembly C5.2 Upper Girth Weld (UGW) C5.3 Upper Axial Weld (UAW)
4.5	Lower Support Structure C5.4 Lower Core Support Beams
4.5	Core Support Barrel Assembly C6.1. Middle Axial Weld (MAW) C6.2. Lower Axial Weld
4.5	Core Shroud Assembly (Welded) C2.1. Remaining Axial Welds
4.5	Core Shroud Assembly (Welded) C3.1. Remaining axial welds
4.6	Control Rod Guide Tube Assembly – W2.1.Remaining CRGT lower flange welds
4.6	Core Barrel Assembly – W3.1.Upper Girth Weld (UGW)
4.6	Core Barrel Assembly – W3.2.Upper Axial Weld (UAW)
4.6	Core Barrel Assembly – W3.3.Lower Flange Weld (LFW)
4.6	Core Barrel Assembly – W4.2. Middle Axial Welds (MAW) W4.3. Lower Axial Welds (LAW)

RAI 21

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

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

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

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

b. Why was the expansion criterion of more than 25% of the bolts on a single plate [degraded] removed in Revision 1 especially considering recent OE with clustered baffle-former bolt degradation?

RAI 22

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

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

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

In MRP-227, Rev. 1, the expansion criteria state:

> Gross cracking (if confirmed) within one inch of a bolt or flow hole location in the baffle plates shall require:
>
> a) An evaluation of the former plates and the core barrel cylinder for the purpose of determining continued operation or repair/replacement by the completion of the next refueling outage. Alternatively, repair/replacement activities may be initiated based on results of a best effort former plate and core barrel cylinder examination.
>
> b) That the VT-3 examination be expanded by the completion of the next refueling outage to include 100% of the accessible portions of the lower grid rib section heat-affected zones adjacent to the in-core monitoring instrumentation (IMI) guide tube spider-to-lower grid rib section welds.

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

The NRC staff therefore requests the following information:

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

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

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

<u>RAI 23</u>

In Table 5-2, “CE Plants Examination Acceptance and Expansion Criteria,” for the Core Shroud Assembly (welded) – Assembly, the examination acceptance criteria in MRP-227, Rev. 1 specifies a VT-3 examination but a VT-1 examination is specified in Table 4-2 for this item. MRP-227-A specified VT-1 in both tables for this item. Clarify whether VT-1 or VT-3 is the intended technique. If VT-3 is the intended technique, explain why this technique is acceptable to address the amount of physical separation expected if distortion is occurring.

<u>RAI 24</u>

In MRP-227, Rev. 1, Table 5-2, the expansion criteria for the UFW requires inspection of the UGW, LGW, and UAW by the completion of the next refueling outage. However, the lower core support beams require inspection within the next three refueling outages. In MRP-227-A, for the corresponding item in Table 5-2, the Core support Barrel Assembly – Upper (core support barrel) flange weld, the expansion to the lower core support beams was required by the completion of the next refueling outage. What is the technical basis for changing the time frame for the expansion inspection of the lower core support beams to within the next three refueling cycles?

RAI 25

In Table 4-1, or Item B15. “IMI Guide Tube Assembly Spiders and Spider welds,” – the examination coverage changed from “100% of top surfaces of 52 spider castings and welds to the adjacent lower grid rib section” in MRP-227-A to “Spiders: 100% of the accessible top surfaces and 100% of the accessible spider surfaces adjacent to the spider casting welds” and “Spider welds: 100% of the accessible welds to the adjacent lower grid rib section.”

The NRC staff requests that EPRI explain why the description of the examination coverage for this item changed, and explain the significance of this change.

RAI 26

For the Westinghouse core barrel assembly, two welds have been reclassified from Primary to Expansion in MRP-227, Rev. 1. The nomenclature has also been changed for some of the welds and some of the weld items in MRP-227-A have been subdivided in MRP-227, Rev. 1. Table 3 below provides the MRP-227-A item name, and the equivalent MRP-227, Rev. 1 item name.

Table 3 – Westinghouse Core Barrel Assembly Weld Items Reclassified from Primary to Expansion

Table	MRP-227-A Primary Item	MRP-227, Rev. 1 Primary Item(s)	MRP-227, Rev. 1 Expansion Item(s)
4.3	Core Barrel Assembly – Upper and Lower Core Barrel Cylinder Girth Welds	Core Barrel Assembly W4. Lower Girth Weld (LGW) (Primary)	Core Barrel Assembly W3.1.Upper Girth Weld (expansion) (UGW)
4.3	Core Barrel Assembly – Lower Core Barrel Flange Weld	n/a	Core Barrel Assembly W3.3.Lower Flange Weld (LFW)

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

In addition, per Table 5-3, the expansion to Table 4.6, Core Barrel Assembly W3.2.Upper Axial Weld (UAW), would only occur if indications are found in either the UGW or the LFW, which are also expansion items. Therefore, it could be as much as four years between the detection of degradation in the primary item until the UAW are examined.

The NRC staff requests that EPRI:

a. Justify reclassifying the UGW and LGW from Primary to Expansion.

b. Justify making the UAW a “secondary expansion” to the UGW and LFW.

c. Justify reclassifying the LFW from Primary to Expansion. Explain why the LFW classification is not consistent with the analogous CE component, the CSBFW, which is classified as Primary.

RAI 27 (RAI No. MRP-227-Rev-1-DLR-2)

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

Section 7.3 in MRP-227, Revision 1, omitted the following information that was previously included in Section 7.3 of the MRP-227-A report:

> Consistent with the requirements of NEI 03-08, if the guidance contained in Table 4-1 through 4-9 and/or Tables 5-1 through 5-3 cannot, need not, or will not be implemented as written, a technical justification must be prepared that clearly states what requirement cannot, need not, or will not be met and why; what alternative action is being taken to satisfy the objective or intent of the guidance; and, why the alternative action is acceptable. Examples of alternatives that may be justifiable are: elevation of an Expansion component to Primary; substitution of an equivalent or more rigorous examination than is required by the tables; or destructive testing in lieu of nondestructive examination, such as the case where one or more of the primary components is being replaced. Since the Expansion components are also “needed” requirements, the technical justification for not fully implementing a Primary component examination or not implementing it in a manner consistent with its intent, would be expected to include disposition of the associated Expansion components.
>
> When submittal of a deviation from work products or elements is required, the justification shall be reviewed and approved in accordance with the applicable plant procedures with the additional responsibility for deviation from a ‘Needed’ element that an internal independent review is performed and that concurrence is obtained from the responsible utility executive. Further, as stipulated in the Implementation Protocol (Appendix B) of NEI 03-08, a utility is required to notify the Issue Program (e.g., the MRP) and the NRC.

Justify the basis for omitting these paragraphs from the scope of Section 7.3 of the MRP-227, Revision 1 report.

REFERENCES

1. EPRI - Report Transmittal: Materials Reliability Program: Pressurized Water Reactor Internals Inspection and Evaluations Guideline (MRP-227, Revision 1), December 21, 2015 (ADAMS Accession No. ML15358A046)

2. Westinghouse Nuclear Safety Advisory Letter (NSAL)-16-1 Revision 1, “Baffle-Former Bolts,” Westinghouse Electric Co. LLC, August 1, 2016.

3. Letter from Bernie Rudell and Anne Demma to the NRC, Subject: “Transmittal of NEI-03-08, ‘Needed’ Interim Guidance Regarding Baffle Former Bolt inspections for Tier 1 plants as Defined in Westinghouse NSAL-16-01,” EPRI Materials Reliability Program, transmitted by letter MRP 2016-022, July 27, 2016 (ADAMS Accession No. ML16211A054).

4. Letter from Bernie Rudell and Brian Burgos to the NRC, Subject: Transmittal of NEI-03-08 “Needed” Interim Guidance Regarding Baffle Former Bolt inspections for Tier 2 plants as Defined in Westinghouse NSAL 16-01, MRP Letter 2017-002, January 12, 2017 (ADAMS Accession No. ML17017A165).

5. WCAP-17096-NP-A, Rev 2, "Reactor Internals Acceptance Criteria Methodology and Data Requirements." August 31, 2016 (ADAMS Accession No. ML16279A320)

6. 1018292, "Materials Reliability Program: Screening, Categorization, and Ranking of B&W-Designed PWR Internals Component Items (MRP-189-Rev. 1)." March 31, 2009 (ADAMS Accession No. ML092230735)

7. 1013233, "Materials Reliability Program: Failure Modes, Effects, and Criticality Analysis of B&W-Designed PWR Internals (MRP-190)." November 30, 2006 (ADAMS Accession No. ML091910128)

8. Revision 1 to the Final Safety Evaluation of Electric Power Research Institute (EPRI) Report, Materials Reliability Program (MRP) Report 1016596 (MRP-227), Revision 0,“Pressurized Water Reactor (PWR) Internals Inspection and Evaluation Guidelines” (TAC No. ME0680), December 16, 2011 (ADAMS Accession No. Ml11308A770).

9. Notification of the Potential Existence of Defects Pursuant to 10 CFR Part 21, Westinghouse Letter LTR-NRC-16-74, November 18, 2016(ADAMS Accession No. ML16328A310).

10. WCAP-17451-P, Rev. 1, "Reactor Internals Guide Tube Wear - Westinghouse Domestic Fleet Operational Projects," October 31, 2013 (ADAMS Accession No. ML15041A107)