January 29, 2018

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

SUBJECT: 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," (CAC NO. MF7223 EPID: L-2016-TOP-0001)

Dear Ms. Burgos:

By letter dated January 12, 2009 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML090160204), the Electric Power Research Institute (EPRI) submitted for U.S. Nuclear Regulatory Commission (NRC) staff review, "TR [Topical Report] Materials Reliability Program (MRP) Report 1016596 (MRP-227), Revision 0, 'Pressurized Water Reactor (PWR) Internals Inspection and Evaluation Guidelines.'" By letter dated December 16, 2011, the NRC staff issued its safety evaluation (SE) on MRP-227, Revision 0. A copy of the NRC staff transmittal letter and SE can be found in ADAMS Accession No. ML11308A770.

By letter dated January 3, 2018 (ADAMS Accession No. ML16238A133), a U.S. Nuclear Regulatory Commission (NRC) draft SE addressing Action Items 1 and 7 from the December 16, 2011, SE was provided for your review and comment. By email dated January 15, 2018 (ADAMS Accession No. ML18016A001), EPRI provided comments on the NRC draft SE. The comments provided by EPRI corrected typographical errors in the SE. The NRC staff's disposition of the EPRI comments on the draft SE is documented in the final SE enclosed with this letter.

Based on the information submitted, the NRC staff has concluded that Action Items 1 and 7 are resolved, and plant-specific responses are generally not necessary, except as delineated in the conclusions found in Section 3.0 of the NRC SE. The final SE defines the basis for our conclusion.

Our acceptance applies only to the material provided. We do not intend to repeat our review of the acceptable material. When the information appears as a reference in a licensing action request, our review will ensure that the material presented applies to the specific plant involved. Requests for licensing actions that deviate from this material 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 approved version of MRP-227 within three months of receipt of this letter. The approved versions shall incorporate this letter and the enclosed final SE after the title page.

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

If you have any questions, please contact the Project Manager for the review, Joseph J. Holonich at 301-415-7297 or by electronic mail at 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 OF ACTION ITEMS 1 AND 7 FROM TOPICAL REPORT MRP-227-A, "MATERIALS RELIABILITY PROGRAM: PRESSURIZED WATER REACTOR INTERNALS INSPECTION AND EVALUATIONS GUIDELINE," (CAC NO. MF7223 EPID: L-2016-TOP-0001) DATED: JANUARY 28, 2018

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U.S. NUCLEAR REGULATORY COMMISSION SAFETY EVALUATION

STATUS OF RESOLUTION OF APPLICANT/LICENSEE ACTION ITEMS 1 AND 7

FROM THE FINAL SAFETY EVALUATION OF MRP-227, REVISION 0

1.0 INTRODUCTION

By letter dated January 12, 2009 (Ref. 1), the Electric Power Research Institute (EPRI), submitted "Materials Reliability Program: Pressurized Water Reactor Internals Inspection and Evaluation Guidelines (MRP-227-Rev. 0)" (Ref. 2) to the Nuclear Regulatory Commission (NRC) for review. In its December 16, 2011, final safety evaluation (SE) of MRP-227, Rev. 0, (Ref. 3), the NRC staff identified eight applicant/licensee action items (A/LAIs) related to issues that could not be resolved on a generic basis.

A/LAI 1 and A/LAI 7 have been the most difficult and time consuming of the A/LAIs for applicants and licensees to resolve. Since the final SE was issued, the EPRI Materials Reliability Program (MRP) and the Pressurized Water Reactor (PWR) Owners Group (PWROG) have completed several projects with the objective of generically resolving A/LAI 1 and A/LAI 7.

The NRC staff final SE is incorporated in the accepted-for-use version of the topical report, MRP-227-A (Ref. 4). The NRC staff has completed its assessment of this generic work and issued several staff assessments on the various industry reports. The following contains the overall NRC SE of the work done to resolve A/LAI 1 and A/LAI 7.

2.0 TECHNICAL EVALUATION

2.1 A/LAI 1 Applicability of Failure Mode Effects and Criticality Analysis and Functionality Analysis Assumptions

A/LAI 1 stated that each applicant/licensee is responsible for assessing its plant's design and operating history and demonstrating that MRP-227-A is applicable to the facility. Each applicant/licensee shall refer, in particular, to the assumptions regarding plant design and operating history made in the failure modes, effects and criticality analysis and functionality analyses for reactors of its design (i.e., Westinghouse Electric Company (Westinghouse), Combustion Engineering (CE), or Babcock and Wilcox (B&W)) which support MRP-227). Also, an applicant/licensee shall describe the process used for determining plant-specific differences in the design of its reactor vessel internals (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 MRP-227-A.

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 (Refs 6, 7, 8, and 9).

To resolve the generic issue of the information needed from licensees to resolve A/LAI 1, a series of proprietary and public meetings were conducted. At the 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. A summary of the proprietary meeting presentations and supporting proprietary generic design basis information is contained in Westinghouse proprietary report WCAP-17780-P (Ref. 10). WCAP-17780-P provides background proprietary design information regarding variances in stress, fluence, and temperature in the plants designed by Westinghouse and CE to support NRC reviews of utility submittals to demonstrate plant-specific applicability of MRP-227-A.

As a result of the technical discussions with the NRC staff, the basis for a plant to respond to the NRC staff request for additional information (RAI) 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 (Refs. 11 and 12):

> 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 [kilo-pounds per square inch]? (If both conditions are true, additional components may need to be screened in for stress corrosion cracking (SCC).
>
> Question 2: Does the plant have an 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 11 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. 13), 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. 14). In the assessment, the NRC staff concluded that if an applicant/licensee demonstrates that its plant(s) comply with the guidance in MRP Letter 2013-025, there is reasonable assurance that the inspection and evaluation 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 acceptable 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, and the information on power uprate effects on fluence and temperature, 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 rather than just stating that the plant complies with the parameter. Those are 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 upper core plate distance for Westinghouse-design reactors.

The guidance in MRP Letter 2013-025 is incorporated into MRP-227, Rev. 1 (Ref. 5) as Appendix B and the updated Section 2.4 of MRP-227, Rev. 1 (Ref. 5) refers to this guidance for Westinghouse and CE plants. EPRI submitted MRP-227, Rev. 1 on December 21, 2015, for staff review (Ref. 5). MRP-227, Rev. 1 is currently under review by the staff.

To generically resolve Question 1 the PWR Owners Group (PWROG) developed PWROG-15105-NP, Revision 0, "PA-MSC-1288 PWR RV Internals Cold-Work Assessment," dated April 30, 2016 (Ref. 15), which was submitted to the NRC for information by letter dated June 15, 2016 (Ref. 16). The NRC staff assessment of PWROG-15105-NP, Revision 0, dated April 21, 2017 (Ref. 17), concluded:

- The majority of austenitic SS materials were required to be solution annealed, which eliminates the possibility of effects from cold work on the SCC behavior of the materials,
- 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,
- No non-fastener RVI components were subject to cold work greater than 20% in PWR units, and these components are less susceptible to SCC.
- 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 acceptably 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 NRC 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 staff recommended in its assessment dated November 7, 2014, that plant-specific values of core-design related parameters be documented and provided. Based on the above, the NRC staff finds that plant-specific applicability of MRP-227, Rev. 1 can be acceptably addressed if an applicant/licensee verifies that its plant meets the criteria of Section 2.4 and Appendix B of MRP-227, Rev. 1.

An applicant/licensee should include this information in its plant-specific RVI program plan. 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, Rev. 1, should be included in the plant-specific RVI program plan.

Based on the above, A/LAI 1 can be resolved by future applicants/licensees submitting plant-specific RVI programs in accordance with MRP-227-A., if the RVI program documents compliance with the criteria of MRP-227-A, Section 2.4, and 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, Rev. 1, are included in the plant-specific RVI program plan.

2.2 A/LAI 7 Plant-Specific Evaluation of Cast Austenitic Stainless Steel 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 [In-core Monitoring Instrumentation] guide tube assembly spiders and CRGT [control-rod guide tube] 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 [cast austenitic stainless steel], 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 staff's SE of MRP-227, Rev. 0 due to concerns about the potential for a synergistic effect of thermal embrittlement (TE) and irradiation embrittlement (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 for MRP-227, Rev. 0, the PWROG issued several technical reports intended to generically address aspects of A/LAI 7.

By letter dated March 13, 2015 (Ref.18), the PWROG submitted to the NRC for information only PWROG-14048-P, Revision 0, "Functionality Analysis: Lower Support Columns" (Ref. 19). PWROG-14048-P, Revision 0 contains a generic methodology for evaluating the functionality of Westinghouse and CE lower support columns (LSCs). The NRC staff assessment of PWROG-14048-P, Revision 0 (Ref. 20) concluded:

- 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.

- The failure tolerance evaluation of the LSCs to demonstrate structural redundancy in the lower support structure (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.

- 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.

By letter dated March 1, 2017 (Ref. 21), the PWROG submitted to NRC for information only PWROG-14048-P, Revision 1, "Functionality Analysis: Lower Support Columns" (Ref. 22). The PWROG revised PWRG-14048-P with the intention of demonstrating that the LSC functionality analysis is generically applicable to all Westinghouse and CE plants participating in the PWROG program. The NRC staff notes that all 48 Westinghouse reactors and all 9 operating CE reactors that have LSCs participated in the program. The NRC staff assessment of PWROG-14048-P, Rev. 1 (Ref. 23) concluded that:

- The flaw tolerance analyses of the four LSC designs representing participating plants demonstrate that the likelihood of full-section failure of LSCs is low.
- 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 acceptably address the range of plant-specific geometric parameters, loading conditions, and acceptance criteria of participating plants; and that the structural redundancy evaluation acceptably included the effect of clusters of failed LSCs.
- The discussion of LSC buckling in the redundancy analysis acceptably addressed buckling of an LSC subject only to a compressive axial load. It did not address the effect of eccentric[1] loading in LSC buckling. The discussion of the LSS assembly dynamic response described in the report acceptably shows there is little change in the dynamic response of the LSS assembly due to failed LSCs.

Although the NRC staff identified in its assessment of PWROG-14048, Rev. 1 that the report did not acceptably address the effect of eccentric loading on LSC bucking, the eccentric loads that could cause buckling of LSCs only occur during the faulted condition. The faulted condition analyzed in PWROG-14048, Rev. 1 is very conservative and an unlikely condition since a loss-of-coolant accident 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

[1] 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.

having LSCs subject to high eccentric loads is low. Therefore, the NRC staff determined there is reasonable assurance that the LSCs, and the LSS, would remain functional under design-basis conditions. No plant-specific analysis is therefore necessary to address the effect of eccentric loads on LSC buckling.

By letter dated January 13, 2016, the PWROG submitted to the NRC for information only PWROG-15032-NP, Revision 0, "PA-MSC-1288 Statistical Assessment of PWR RV Internals CASS Materials" (Ref. 24). The report described 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.

PWROG-15032-NP, Revision 0 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. 25) concluded that the report can be used by applicants/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 certified material test reports (CMTRs). These estimated ferrite values may then be used to screen the CASS material for TE. The NRC staff assessment also stated that for CASS components subject to neutron fluences greater than $1x10^{17}$ neutrons per square centimeter (n/cm^2), additional adjustments for the effects of irradiation must be applied to the methodology used to estimate toughness.

In addition, the NRC staff 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 final SE for "BWRVIP-234: Thermal Aging and Neutron Embrittlement Evaluation of Cast Austenitic Stainless Steel for BWR Internals" (Ref. 26). 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 displacements 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 Grimes Letter, (Grimes, Christopher I., Branch Chief, NRC, letter to Douglas J. Walters, NEI, "License Renewal Issue No. 98-0030, 'Thermal Aging Embrittlement of Cast Austenitic Stainless Steel Components,'" May 19, 2000, ADAMS Accession No. ML003717179).

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 American Society of Mechanical Engineers Code, Section XI, "In-service Inspection" 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 core support columns (CSC) in some plants. PWROG-14048-P 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 acceptably 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 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 acceptably managed. Therefore, for all Westinghouse CASS RVI components that are not “No Additional Measures” components, loss of fracture toughness has been acceptably addressed.

B&W RVI generic components fabricated from CASS or precipitation-hardened SS 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 SS). With respect to B&W components, the NRC staff determined that A/LAI 7 is resolved for two licensees (four units) based on functionality evaluations submitted by these licensees, as documented in References 6 and 7. Another licensee has submitted its evaluation (Ref. 27), but the NRC staff has not completed review of this submittal.

One other licensee made a license renewal 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. 8). It is expected that A/LAI 7 will be addressed for the two remaining licensees of B&W units by the submittals. A generic report has not been developed to resolve A/LAI 7 for B&W-design RVI. Since A/LAI 7 has not been resolved generically, the one remaining licensee of a B&W unit that has not submitted its A/LAI 7 evaluation should plan to do so.

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

A/LAI 2 essentially requires that an applicant/licensee check the generic RVI components lists in MRP-189, Rev. 1, “Materials Reliability Program: Screening, Categorization, and Ranking of B&W-Designed PWR Internals Component Items” (for B&W RVI, Ref. 28)) and MRP-191, “Materials Reliability Program: Screening, Categorization, and Ranking of Reactor Internals Components for Westinghouse and Combustion Engineering PWR Design” (for CE and Westinghouse RVI, Ref. 29) against the RVI components that are in scope of license renewal for its plant, and determine if there are any plant-specific RVI components at its plant. If plant-specific components are found, A/LAI 2 requires the applicant/licensee to propose modifications to the MRP-227-A generic program to manage aging of the plant-specific components.

It is possible that plant-specific CASS components could be identified in applicant/licensee responses to A/LAI 2. If these components are not classified as “No Additional Measures” components, then appropriate aging management activities must be identified for the components per A/LAI 2. The ferrite content of these components may be estimated using the

methodology and information in PWROG-15032-NP (Ref. 24). 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, dated June 22, 2016.

3.0 CONCLUSIONS

The NRC staff has reviewed and issued assessments of several industry technical reports submitted to the NRC for information only. The reports were provided in support of generically resolving A/LAI 1 and A/LAI 7 from the NRC staff SE of MRP-227, Rev. 0. Based on its review, the NRC staff concludes:

- A/LAI 1 has been resolved for all operating B&W reactors.

- A/LAI 1 can be resolved by future applicants/licensees of CE or Westinghouse plants submitting plant-specific RVI programs in accordance with MRP-227-A, if the RVI program documents compliance with the criteria of MRP-227-A, Section 2.4, and 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, Rev. 1, are included in the plant-specific RVI program plan.

- A/LAI 7 has been acceptably resolved for 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 been resolved on a plant-specific basis for four of six operating B&W reactors. Licensees of plants with B&W-design RVI should submit plant-specific A/LAI 7 evaluations if they have yet to do so.

- Aging management of any plant-specific CASS components may be addressed in the response to A/LAI 2. For plant-specific CASS components for which aging management activities are necessary under A/LAI 2, the ferrite content of these components may be estimated using the methodology and information in PWROG-15032-NP (Ref. 24). Screening of these components for IE and TE may be done according to the criteria found in Appendix B to the Final SE of BWRVIP-234, dated June 22, 2016 (Ref. 26).

4.0 REFERENCES

1. Report Transmittal; Materials Reliability Program: Pressurized Water Reactor Internals Inspection and Evaluation Guidelines (MRP-227-Rev. 0), January 12, 2009 (ADAMS Accession No. ML090160204)

2. EPRI, “Materials Reliability Program: Pressurized Water Reactor Internals Inspection and Evaluation Guidelines (MRP-227-Rev. 0),” Report 1016596, Palo Alto, CA, December 2008 (ADAMS Accession No. ML090160205)

3. 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. MI11308A770)

4. EPRI, "Materials Reliability Program: Pressurized Water Reactor Internals Inspection and Evaluation Guidelines (MRP-227-A)," Report 1022863, Palo Alto, CA, December 2011 (ADAMS Package Accession No. ML120170453)

.

5. 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)

6. 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)

7. 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 19, 2015 (ADAMS Accession No. ML15050A671)

8. 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)

9. 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)

10. Westinghouse Report, WCAP-17780-P, Rev. 0, "Reactor Internals Aging Management MRP-227-A Applicability for Combustion Engineering and Westinghouse Pressurized Water Reactor Designs," June 2013 (PROPRIETARY – NOT PUBLICALLY AVAILABLE)

11. NRC Letter, "Summary of May 21, 2013, Public Meeting Regarding Pressurized Water Reactor (PWR) Vessel Internals Inspections," June 24, 2013 (ADAMS Accession No. ML13164A126)

12. NRC Presentation, "Status of MRP-227-A Action Items 1 and 7," June 5, 2013 (ADAMS Accession No. ML13154A152)

13. EPRI, MRP-227-A Applicability Template Guideline, MRP Letter 2013-025, October 14, 2013 (ADAMS Accession No. ML13322A454)

14. 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 And Westinghouse Pressurized Water Reactor Designs, November 7, 2014

15. (ADAMS Accession No. ML14309A484)

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17. 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)

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19. 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)

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

21. Summary Assessment Of PWROG-14048-P, Revision 0, "Functionality Analysis: Lower Support Columns," December 17, 2017 (ADAMS Accession No. ML15334A462)

22. 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)

23. PWROG-14048-P, Revision I, "Functionality Analysis: Lower Support Columns," February 28, 2017 (ADAMS Accession No. ML17066A267)

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

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26. 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)

27. Letter and Final BWRVIP-234 Safety Evaluation, Thermal Aging and Neutron Embrittlement Evaluation of Cast Austenitic Stainless Steel for BWR Internals TAC No. ME5060, June 22, 2016 (ADAMS Accession No. ML16096A002)

28. Three Mile Island, Unit 1, "MRP-227-A Applicant/Licensee Action Item 7 Analysis, Topical Report," ANP-3479NP, Revision 0," August 31, 2016 (ADAMS Accession No. ML16263A319)

29. Final Report, MRP-189, Rev. 1, "Materials Reliability Program: Screening, Categorization, and Ranking of B&W-Designed PWR Internals Component Items." March 31, 2009 (ADAMS Accession No. ML091671778)

30. 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)

Attachment: Resolution of Comments

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Date: January 29, 2018

Resolution of Comments

Location	Comment	NRC Staff Response
Page 1, Section 1.0, 3rd paragraph	Rev. 4 should be Ref. 4	Agreed
Page 1, Section 2.1, Header	Effect should Effects	Agreed
Page 3, Section 2.1, 2nd paragraph	"The guidance in MRP Letter 2013-025 is incorporated as Appendix B. Section 2.4 of MRP-227, Rev. 1 (Ref. 5) refers to this guidance for Westinghouse and CE plants." Should be '…as Appendix B. ~~Section 2.4~~ of MRP-227, Rev. 1 (Ref. 5)…'	Agreed
Page 8, Section 3.0, 1st paragraph, 2nd Sentence	"There" should be "The."	Agreed

Attachment