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

October 7, 2016

Mr. Eric A. Larson, Site Vice President
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Beaver Valley Power Station
Mail Stop A-BV-SEB1
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SUBJECT: BEAVER VALLEY POWER STATION, UNIT NOS. 1 AND 2 – STAFF ASSESSMENT OF THE REACTOR VESSEL INTERNALS AGING MANAGEMENT PROGRAM PLANS (CAC NOS. MF3416 AND MF3417)

Dear Mr. Larson:

By letter dated January 27, 2014, as supplemented by letters dated August 26, 2015, and November 6, 2015, FirstEnergy Nuclear Operating Company (the licensee), submitted Reactor Vessel Internals (RVI) Aging Management Program Plans (AMPs) for the Beaver Valley Power Station, Unit Nos. 1 and 2 (BVPS-1 and BVPS-2). The RVI AMPs, including RVI inspection plans, were submitted to fulfill License Renewal Commitment No. 18 for BVPS-1 and Commitment No. 20 for BVPS-2, as documented in Appendix A of NUREG-1929, "Safety Evaluation Report Related to the License Renewal of Beaver Valley Power Station, Units 1 and 2." The RVI AMPs are based on "Materials Reliability Program: Pressurized Water Reactor Internals Inspection and Evaluation Guidelines (MRP-227-A)." License Renewal Commitment No. 18 for BVPS-1 and Commitment No. 20 for BVPS-2 were fulfilled upon submittal of the RVI AMPs on January 27, 2014.

The U.S. Nuclear Regulatory Commission (NRC) staff's review of the licensee's RVI AMPs is provided in the enclosed staff assessment. The NRC staff concludes that the licensee's RVI AMPs are acceptable, because they are consistent with the inspection and evaluation guidelines of MRP-227-A, and the licensee adequately addressed all eight licensee action items specified in MRP-227-A.

The NRC staff's approval of the BVPS-1 and BVPS-2 RVI AMPs does not reduce, alter, or otherwise affect current American Society of Mechanical Engineers Code, Section XI, Inservice Inspection (ISI) requirements, or any BVPS-1 and BVPS-2 specific licensing requirements related to ISIs.

If you have any questions concerning this matter, please contact the Project Manager, Taylor Lamb, at (301) 415-7128 or Taylor.Lamb@nrc.gov.

Sincerely,

for Douglas A. Broaddus, Chief
Plant Licensing Branch I-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. 50-334 and 50-412

Enclosure:
Staff Assessment

cc w/enclosure: Distribution via Listserv

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STAFF ASSESSMENT BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO THE AGING MANAGEMENT PROGRAM PLANS

FOR REACTOR VESSEL INTERNALS

FIRSTENERGY NUCLEAR OPERATING COMPANY

BEAVER VALLEY POWER STATION, UNIT NOS. 1 AND 2

DOCKET NOS. 50-334 AND 50-412

1.0 INTRODUCTION AND BACKGROUND

By letter dated January 27, 2014 (Reference 1), as supplemented by letters dated August 26, 2015, and November 6, 2015 (Reference 2 and Reference 3, respectively), FirstEnergy Nuclear Operating Company (FENOC, the licensee), submitted inspection plans for reactor vessel internals (RVI) for the Beaver Valley Power Station, Unit Nos. 1 and 2 (BVPS-1 and BVPS-2) (Reference 4 and Reference 5, respectively). Since the inspection plans are intended to manage the effect of aging on passive long-lived structures or components, this type of inspection plan is commonly referred to as an aging management program. The Materials Reliability Program (MRP)-227-A Topical Report (TR) (Reference 6) was used by the licensee as the technical basis for developing the BVPS-1 and BVPS-2 AMPs. The licensee submitted the RVI AMPs to fulfill a license condition that required the completion of license renewal commitments, which include License Renewal Commitment No. 18 for BVPS-1 and Commitment No. 20 for BVPS-2 from Appendix A of NUREG-1929, "Safety Evaluation Report Related to the License Renewal of Beaver Valley Power Station, Units 1 and 2" (Reference 7). The RVI AMPs include inspection and evaluation (I&E) guidelines for the RVI components at BVPS-1 and BVPS-2.

2.0 REGULATORY EVALUATION

Each of the renewed licenses for BVPS-1 and BVPS-2 contain a license condition concerning license renewal commitments. The license conditions elevate the regulatory commitments provided by FENOC during its application for renewed licenses for BVPS-1 and BVPS-2 into requirements.

BVPS-1 License Condition 2.F states the following:

> License Renewal Commitments – The UFSAR supplement, as revised, submitted pursuant to 10 CFR 54.21(d), describes certain future activities to be completed prior to and/or during the period of extended operation. FENOC shall complete these activities in accordance with Appendix A of NUREG-1929, Safety Evaluation Report Related to the Beaver Valley Power Station, Units 1 and 2, dated October 2009, and Supplement 1 of NUREG-1929, dated October 2009,

> and shall notify the NRC in writing when activities to be completed prior to the period of extended operation are complete and can be verified by NRC inspection.

Appendix A, "BVPS Units 1 and 2 License Renewal Commitments," of NUREG-1929, contains commitments associated with the license renewal of BVPS-1 and BVPS-2. Commitment No. 18 states, in part:

> [...] regarding activities for managing the aging of Reactor Vessel internal components and structures, BVPS commits to: 1. Participate in the industry programs applicable to BVPS Unit 1 for investigating and managing aging effects on reactor internals; 2. Evaluate and implement the results of the industry programs as applicable to the BVPS Unit 1 reactor internals; and, 3. Upon completion of these programs, but not less than 24 months before entering the period of extended operation, submit an inspection plan for the BVPS Unit 1 reactor internals to the NRC for review and approval.

The period of extended operation (PEO) for BVPS-1 began on January 29, 2016. Accordingly, the licensee submitted its RVI inspection plan prior to 24 months before entering the PEO. The licensee stated that the "PWR Vessel Internals Program Plan for Aging Management of Reactor Internals at Beaver Valley Power Station Unit 1" is based upon License Renewal Commitment No. 18 and MRP-227-A.

BVPS-2 License Condition 2.I states the following:

> License Renewal Commitments – The UFSAR supplement, as revised, submitted pursuant to 10 CFR 54.21(d), describes certain future activities to be completed prior to and/or during the period of extended operation. FENOC shall complete these activities in accordance with Appendix A of NUREG-1929, Safety Evaluation Report Related to the Beaver Valley Power Station, Units 1 and 2, dated October 2009, and Supplement 1 of NUREG-1929, dated October 2009, and shall notify the NRC in writing when activities to be completed prior to the period of extended operation are complete and can be verified by NRC inspection.

BVPS-2 Commitment No. 20 states, in part:

> ... regarding activities for managing the aging of Reactor Vessel internal components and structures, BVPS commits to: 1. Participate in the industry programs applicable to BVPS Unit 2 for investigating and managing aging effects on reactor internals; 2. Evaluate and implement the results of the industry programs as applicable to the BVPS Unit 2 reactor internals; and 3. Upon completion of these programs, but not less than 24 months before entering the period of extended operation, submit an inspection plan for the BVPS Unit 2 reactor internals to the NRC for review and approval.

The PEO for BVPS-2 will begin on May 27, 2027. Accordingly, the licensee submitted its RVI inspection plan prior to 24 months before entering the PEO. The licensee stated that the "PWR Vessel Internals Program Plan for Aging Management of Reactor Internals at Beaver Valley Power Station Unit 2" is based upon License Renewal Commitment No. 20 and MRP-227-A.

The MRP-227-A TR and MRP-232, which reference MRP-191 (References 20 and 21), comprise an industry program that is applicable to the BVPS-1 and BVPS-2 reactor internals. On January 12, 2009, the Electric Power Research Institute (EPRI) submitted for U.S. Nuclear Regulatory Commission (NRC) staff review and approval the MRP-227 Report, Revision 0 (Reference 8), which was intended as guidance for applicants in developing their plant-specific AMPs for RVI components. The final NRC safety evaluation (SE) regarding MRP-227, Revision 0 (Reference 9), was issued on December 16, 2011. This SE, which has been included in MRP-227-A, contains specific conditions on the use of the TR and applicant/licensee action items (AIs) that must be addressed by those utilizing the TR as the basis for a submittal to the NRC. On January 9, 2012, EPRI published the NRC-approved version of the TR, designated MRP-227-A. MRP-227-A contains a discussion of the technical basis for the development of plant-specific AMPs for RVI components in pressurized-water reactor (PWR) vessels and also provides I&E 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 the effects of aging on RVI components.

The scope of components considered for inspection under the guidance of MRP-227-A includes core support structures, which are typically denoted as Examination Category B-N-3 by Section XI of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code), and those RVI components that serve an intended function consistent with the criteria in 10 CFR 54.4(a)(1). The scope of the program does not include short-lived or active components such as fuel assemblies, reactivity control assemblies, and nuclear instrumentation, because these components are not subject to an aging management review (AMR), as defined in 10 CFR 54.21(a)(1).

MRP-232 summarizes the development of an aging management strategy for Westinghouse and Combustion Engineering (CE) reactor internals. This report provides the technical basis for the aging management requirements of Westinghouse and CE reactor internals. MRP-191 describes the process and results of categorizing Westinghouse and CE-designed PWR internals components according to age-related degradation and significance. These results are a key element in developing inspection and evaluation guidelines for aging management of PWR internals. In addition to the MRP reports, WCAP-17451-P and WCAP-17780-P provide information that can aid implementation of industry programs. WCAP-17451-P provides projections for reactor internals guide tube wear for the Westinghouse domestic fleet. WCAP-17780-P provides a summary of the background technical basis for the broad applicability of the MRP-227-A guidelines.

Subsequent to the submittal of MRP-227, NUREG-1801, Revision 2, "Generic Aging Lessons Learned (GALL) Report – Final Report" (Reference 10), was issued, providing new AMR line items and aging management guidance in AMP XI.M16A, "PWR Vessel Internals." This AMP was based on staff expectations for the guidance to be provided in MRP-227-A. Since the GALL was published prior to the issuance of the final SE on MRP-227-A, the NRC staff published "Final License Renewal Interim Staff Guidance, LR-ISG-2011-04, Updated Aging

Management Criteria for Reactor Vessel Internal Components for Pressurized Water Reactors" (Reference 11), which modifies the guidance of AMP XI.M16A to be consistent with MRP-227-A.

3.0 TECHNICAL EVALUATION

The NRC staff's assessment of the BVPS-1 and BVPS-2 RVI AMPs focused on determining whether the licensee incorporated the generic I&E guidelines recommended in MRP-227-A into the plant-specific AMPs. Therefore, the dominant aspects of the staff's assessment focused on the I&E guidelines specified in the BVPS-1 and BVPS-2 RVI AMPs (and items relevant to these guidelines) and the licensee's responses to the AIs from the NRC staff's SE for MRP-227-A. Specifically, the NRC staff's assessment focused on the following: (1) the licensee's implementation of the inspection program in MRP-227-A for RVI components in the primary, expansion, and existing categories, as well as the appropriate acceptance criteria; (2) operating experience at BVPS-1 and BVPS-2; (3) the licensee's resolutions of the eight AIs in the NRC staff's SE for MRP-227-A; and (4) the licensee's responses to the TR conditions addressed in the NRC staff's SE for MRP-227-A. The following subsections provide details of the NRC staff's assessment of these items.

3.1 Assessment of Licensee's Implementation of Inspection Program for RVI Components in the Primary, Expansion, and Existing Categories, and Acceptance Criteria

The licensee implemented the MRP-227-A inspection program in the primary, expansion, and existing categories in the tables in Enclosures A and B of the submittal: Table C-1, "MRP-227-A Primary Inspection and Monitoring Recommendations for Westinghouse-Designed Internals"; Table C-2, "MRP-227-A Expansion Inspection and Monitoring Recommendations for Westinghouse-Designed Internals"; and Table C-3, "MRP-227-A Existing Inspection and Aging Management Recommendations for Westinghouse-Designed Internals." The staff reviewed these tables and determined that they are consistent with Table 4-3, "Westinghouse Plants Primary Components"; Table 4-6, "Westinghouse Plants Expansion Components"; and Table 4-9, "Westinghouse Plants Existing Programs Components," respectively, of MRP-227-A.

Additionally, the staff reviewed acceptance criteria in Table C-4, "MRP-227-A Acceptance Criteria and Expansion Criteria Recommendations for Westinghouse-Designed Internals," of Attachments 1 and 2 of the licensee's submittal and determined that it is consistent with Table 5-3, "Westinghouse Plants Examination Acceptance and Expansion Criteria," of MRP-227-A.

3.2 Assessment of the Licensee's AMP for RVI Components - Operating Experience

The NRC staff reviewed the program elements for the licensee's AMP against those found in Section A.1.2.3 and Table A.1-1 of NUREG-1800, Revision 2, "Standard Review Plan for Review of License Renewal Applications for Nuclear Power Plants" (Reference 12). Based on review of the AMP for each unit in the following subsections, the NRC staff finds the program elements of BVPS-1 and BVPS-2 AMPs meet the recommended criteria for these elements in Section A.1.2.3 of NUREG-1800, Revision 2.

3.2.1 Control Rod Guide Tube Support Pins and Flux Thimble Tubes

In Section 5.10, "GALL Revision 2 Element 10: Operating Experience," the licensee summarizes the plant-specific operating experience (OE) for each unit. Only two significant pieces of OE for BVPS were noted: control rod guide tube (CRGT) support pins and flux thimble tubes. In each unit, the licensee proactively replaced the existing CRGT support pins and plugged the flux thimble tubes that were projected to approach the acceptance criteria for wall thinning under an approved existing AMP. The AMP for loss of material due to wear in the flux thimble tubes provided in Section B.2.19 of the BVPS license renewal application was previously approved in Section 3.0.3.2.7 of NUREG-1929. The NRC staff reviewed the OE and concludes that the plant-specific OE is consistent with the generic OE found in MRP-227-A. The licensee did not identify any cracked baffle/former bolts that were discovered at other Westinghouse-designed domestic PWRs.

3.2.2 Control Rod Guide Tube Guide Cards

In Section 5.5, "GALL Revision 2 Element 5: Monitoring and Trending," of the licensee's submittal, it was noted that the Pressurized-Water Reactor Owners Group (PWROG) has recently developed initial examination period requirements for guide plate card wear for Westinghouse-designed plants that replace the current requirements in MRP-227-A. The NRC staff notes that the licensee is planning to replace the requirements of MRP-227-A to inspect all guide card assemblies for both units, based on the recommendations in WCAP-17451-P, Revision 1, "Reactor Internals Guide Tube Wear – Westinghouse Domestic Fleet Operational Projections" (Reference 13). The new guidance would require inspections of the guide cards at both units for wear during an outage within a time range from 30 to 38 effective full-power years, because both units were evaluated as part of the technical basis for WCAP-17451-P, Revision 1. This report provides inspection guidelines and acceptance criteria for monitoring wear in guide cards.

Based on the licensee's evaluation, the staff concludes that the AMP for guide cards is adequately managed at both BVPS units. The staff's conclusion is based on the following. (1) the licensee inspects all guide card assemblies for both units, based on the recommendations in WCAP-17451-P, Revision 1, (2) the licensee would use the fleet-wide inspection results in establishing the subsequent inspection frequency and, (3) if any unacceptable wear in guide cards were to be observed during the future inspections, the licensee would follow the inspection guidelines addressed in WCAP-17451-P, Revision 1.

3.3 <u>Assessment of the Licensee's Resolutions to the Eight Applicant/Licensee Action Items in Section 4.2 of the SE for MRP-227A</u>

3.3.1 Assessment of Resolution to AI-1

The purpose of AI-1 is to assure that the applicant has assessed its plant-specific design and operating history and has determined whether MRP-227-A is applicable to its facility. AI-1 is described in Section 4.2.1 of the SE on MRP-227-A, Revision 0, which states:

> Each applicant/licensee should refer, in particular, to the assumptions regarding plant design and operating history made in the [failure modes, effects and

criticality analysis (FMECA)] and functionality analyses for reactors of their design (i.e., Westinghouse, [CE], or [Babcock and Wilcox (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 MRP provided the Westinghouse Proprietary Report WCAP-17780-P, "Reactor Internals Aging Management MRP-227-A Applicability for Combustion Engineering and Westinghouse Pressurized Water Reactor Designs" (Reference 14), to address information needed from licensees to resolve AI-1. The report includes background proprietary design information regarding variances in stress, fluence, and temperature in the plants designed by Westinghouse and CE. The report was written to support NRC reviews of licensee submittals to demonstrate plant-specific applicability of MRP-227-A.

The MRP developed MRP Letter 2013-025 (Reference 15), dated October 14, 2013, to serve as a checklist of plant-specific information that would allow any plant to demonstrate compliance with the basic applicability assumptions in MRP-227-A. The NRC staff reviewed MRP Letter 2013-025 and the technical basis for the guidance contained in WCAP-17780-P. The staff concluded in its evaluation of WCAP-17780-P (Reference 16) that if an applicant or licensee demonstrates compliance with the guidance in MRP Letter 2013-025, there is reasonable assurance that the I&E guidance of MRP-227-A will be applicable to the specific plant(s).

The generic guidance in MRP 2013-025 provides an acceptance basis for the licensee to respond to the following two issues: (1) effect of cold work in enhancing the susceptibility to SCC and (2) effect of fuel design or fuel management that could render the assumptions of MRP-227-A regarding core loading design non-representative for that plant. These two issues are addressed in the following two questions, respectively. In order to demonstrate that a plant is bounded by the MRP-227-A evaluation for originally licensed and uprated conditions, the licensee would need to provide a satisfactory response to two questions for each Westinghouse unit that had not implemented an extended power uprate after 2007. BVPS-1 and BVPS-2 are in this category.

Question 1: Does the plant have non-weld or non-bolting austenitic stainless steel components with ≥ 20 percent cold work, and, if so, do the affected components have operating stresses greater than 30 kilopound 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 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?

For Question 1, the licensee stated in its August 20, 2015, supplement that BVPS-1 and BVPS-2 do not have non-weld or non-bolting austenitic stainless steel RVI components with ≥ 20 percent cold work. For Question 2, the licensee compared the core geometries and operating characteristics for the two units to the applicability guidelines from MRP 2013-025 and determined that neither unit has utilized atypical fuel designs or fuel management that could invalidate the assumptions of MRP-227-A regarding core loading and core design, including power changes and uprates, over their operating lifetimes. Furthermore, the licensee confirmed that BVPS-1 and BVPS-2 will continue to comply with these limits during the PEO.

The NRC staff requested clarification from the licensee to provide the basis for the plant-specific values for heat generation figure of merit, core power density, and distance between the top of the active fuel and the upper core plate. The licensee provided a supplement dated November 6, 2015, that stated the evaluation was performed consistent with MRP 2013-025, which includes the AMRs conducted as part of license renewal. Specifically, the following actions were identified:

1. Confirm that plant-specific components identified for aging management were included in the MRP-191 (Reference 17) component reviews.

2. Confirm that the design and operating history of those components are consistent with MRP-191.

3. Confirm that modifications or plant-specific activities performed on the component do not introduce cold-worked conditions. Actions to consider for focused assessment of potential impact may include, but are not limited to, annealing, cold bending, or surface grinding.

The November 6, 2015. supplement also stated that the plant-specific values for heat generation figure of merit, core power density, and distance between the top of the active fuel and the upper core plate were all acceptable per the checklist in MRP 2013-025.

The NRC staff reviewed the November 6, 2015, supplement and concluded that the AMRs completed during license renewal and the specific actions performed consisted with MRP 2013-025 provide reasonable assurance that neither BVPS-1 nor BVPS-2 have any non-weld or non-bolting austenitic stainless steel RVI components fabricated from material with ≥ 20 percent cold work. Furthermore, the licensee demonstrated that neither unit utilized atypical fuel designs or fuel management as defined in MRP 2013-025, which could invalidate the assumptions of MRP-227-A.

The NRC staff concludes that the licensee complied with the guidelines related to applicability of MRP-227-A and that implementation of I&E guidelines addressed in MRP-227-A would be valid during the PEO. Therefore, the NRC staff finds that the applicant has adequately addressed AI-1.

3.3.2 Assessment of Resolution to AI-2

AI-2 requires the licensee to perform a plant-specific evaluation, identifying which RVI components are within the scope of license renewal at BVPS-1 and BVPS-2. In addition, the purpose of AI-2 is to verify that the licensee reviewed the information in Tables 4-4 and 4-5 in MRP-191 and identified whether these tables contain all of the RVI components that are within the scope of license renewal for its facility, and if the tables do not identify all the RVI components that are in the scope of license renewal, the licensee shall identify the missing component(s) and propose any modifications to the program as defined in MRP-227-A.

The licensee determined that the generic scoping and screening criteria as summarized in MRP-191 and MRP-232, "Materials Reliability Program: Aging Management Strategies for Westinghouse and Combustion Engineering PWR Internals, 1016593, Final Report, December 2008" (Reference 18), are applicable to the plant-specific RVIs for both units with no modifications. MRP-232 (which references MRP-191) provides technical bases for developing the inspection categories for various RVI components based on their susceptibility to aging degradation, which depends on the type of material used in RVI component.

The NRC staff reviewed the response for AI-2 in the AMPs from the January 27, 2014, submittal, along with the 2007 license renewal application and the November 6, 2015, supplement. The licensee performed the initial AMR as part of license renewal according to GALL, Revision 1, which was approved in NUREG-1929, and the licensee identified some components that were different from GALL, Revision 1. For license renewal, these components were noted to be consistent with GALL for material, environment, and aging effect (referred to as Note C in the AMR tables) and are, therefore, acceptable. The plant-specific components and materials were consistent with the typical Westinghouse RVI components in MRP-191, with the exception of some materials addressed below.

In some cases, the material type for a given plant-specific component was identified as either wrought austenitic stainless steel or cast austenitic stainless steel (CASS). While performing the generic FMECA, MRP had conservatively considered most of these components to be either CASS or wrought material. Therefore, additional plant-specific evaluation that is relevant to CASS materials was limited to only those plant-specific components that were not generically considered as CASS (lower support columns), which are discussed in more detail in Section 3.3.7 of this staff assessment. In MRP-191, these components are not subject to any aging mechanisms related to cracking. Therefore, adding potential susceptibility to loss of fracture toughness due to thermal aging will not change the FMECA results.

The NRC staff concludes that the generic scoping and screening of the RVI components are applicable to BVPS-1 and BVPS-2. The NRC staff concludes that BVPS-1 and BVPS-2 comply with AI-2.

3.3.3 Assessment of Resolution to AI-3

AI-3 in the NRC staff's SE for MRP-227-A requires the licensee to perform a plant-specific evaluation of its existing programs on CRGT support pins at BVPS-1 and BVPS-2. The licensee addressed the CRGT support pins under OE for the AMP, Section 5.10. The original alloy X-750 pins were replaced (in 2007 for BVPS-1 and in 2008 for BVPS-2) at Westinghouse's recommendation.

The NRC staff reviewed the response for AI-3 in the January 27, 2014, submittal and noted that in response to the industry concern for SCC of the alloy X-750, the licensee replaced all of the alloy X-750 CRGT support pins (split pins) with Westinghouse-supplied cold-worked Type 316 stainless steel to mitigate the possibility of SCC of these components. Alloy X-750 is a nickel-base alloy, which is more susceptible to SCC than iron-based alloy (i.e., Type 316 stainless steel). Operating experience to date indicates that Type 316 RVI components have not exhibited SCC in the PWR environment. The CRGT support pins are not part of ASME Section XI inspections, but degradation of original alloy X-750 pins was found by inspections under ASME Section XI. Given the licensee's replacement of alloy X-750 CRGT support pins at BVPS-1 and BVPS-2 with cold-worked Type 316 pins, the staff concludes that the licensee addressed AI-3 satisfactorily.

3.3.4 Assessment of Resolution to AI-4 and AI-6

AI-4 and AI-6 of the staff's SE for MRP-227-A are applicable to RVI components designed by Babcock and Wilcox and, therefore, are not applicable to BVPS-1 and BVPS-2.

3.3.5 Assessment of Resolution to AI-5

AI-5 requires that Westinghouse units identify the plant-specific acceptance criteria to be applied when performing physical measurements for loss of compressibility due to loss of preload for the hold-down springs.

The licensee stated in Section 6.2.5 of the AMPs that BVPS-1 and BVPS-2 utilize Type 304 hold-down springs. Loss of preload is applicable to a more susceptible material such as Type 304 stainless steel. The licensee, in Table 7-1 of the AMPs for BVPS-1 and BVPS-2, stated that it has a commitment to develop acceptance criteria for the physical measurements of hold-down springs in BVPS units.

The NRC staff reviewed the response for AI-5 and notes that the licensee is following the inspection guidelines addressed in Table 4-3 of MRP-227-A.

3.3.6 Assessment of Resolution to AI-7

AI-7 requires a licensee to perform a plant-specific analysis demonstrating that the functionality of the set of components made of potentially susceptible materials (CASS, martensitic stainless steel, or precipitation hardened stainless steel) is maintained during the license renewal period. From MRP-227-A for Westinghouse-designed units, the only generic items related to AI-7 are the lower support columns (LSCs), which for BVPS-1 were identified as CASS in Table 6-2, "Summary of BV Unit 1 CASS Components and their Susceptibility to TE," in Enclosure A of the submittal. The LSCs of BVPS-2 were not one of the components identified as CASS in Table 6-2, "Summary of BV Unit 2 CASS Components and their Susceptibility to TE," in Enclosure B of the submittal. Therefore, only the BVPS-1 LSCs are made of CASS.

The licensee provided evaluations of those components identified in response to AI-2 that were fabricated from CASS materials. For those plant-specific CASS RVI components not generically analyzed as CASS (and susceptible to thermal embrittlement (TE) and irradiation embrittlement (IE)) in MRP-191, the licensee stated that FMECA review was performed. The licensee stated that the LSCs from BVPS-1 are the only components for which further analysis is being pursued under AI-7. The licensee also stated in its submittals that no martensitic stainless steel or martensitic precipitation hardened stainless steel materials were identified in the RVIs for either of the units.

The NRC staff reviewed the licensee's response to AI-7 and noted from the review of AI-2 in Section 3.4.2 of this staff assessment that no change to the MRP-227-A guidelines is needed for implementation at BVPS. In addition, the licensee noted that the LSCs at BVPS-2 are made of wrought (non-cast) stainless steel and, therefore, they are not applicable to AI-7. The only components not generically considered as CASS in MRP-191 (the upper instrumentation supports, brackets, and clamps) are not subject to significant neutron exposure or any aging mechanisms related to cracking. The LSCs from BVPS-1 are the only components fabricated from a potentially susceptible material that requires aging management. In Table 6-2 of the AMP for BVPS-1, the licensee stated that the ferrite content of the cast columns was < 20 percent, so it is not considered to be susceptible to TE. GALL AMP XI.M12, "Thermal Aging Embrittlement of Cast Austenitic Stainless Steel," states that CASS materials with ferrite content less than 20 percent are not susceptible to TE. This is the staff's position and was established in an NRC letter dated May 19, 2000 to NEI regarding License Renewal Issue No. 98-0030 (Reference 19).

In its supplement dated August 26, 2015 (Reference 2), the licensee provided the ferrite content of the LSC bodies for BVPS-1 to determine its susceptibility to TE and justification for the adequacy of the primary inspection component links for the LSC bodies for BVPS-1 and BVPS-2 as an approach to manage the aging of the LSCs (demonstrating their functionality during the license renewal period) if the LSCs are not susceptible to TE. The LSCs for BVPS-1 are fabricated from ASTM A-351, Grade CF-8 material. The ferrite content is calculated based upon the chemical compositions using Hull's equivalent factors method. Of the 68 LSCs in BVPS-1, 54 have calculated ferrite contents ranging from a minimum of 2.6 percent to 15 percent, and 14 had a calculated ferrite content of 15.7 percent.

The NRC staff reviewed the supplement dated August 26, 2015, which follows the guidance from the GALL AMP XI.M12, as stated above, and the NRC May 19, 2000, letter. Since the LSCs at BVPS-1 do not have ferrite content greater than 20 percent, they are not susceptible to TE. The portion of the LSCs in contact with the lower core plate will see an end-of-life fluence that approaches 15 displacements per atom and, therefore, the LSCs are susceptible to IE. The staff determined that in the absence of TE, the combined effect of TE and IE in the LSCs is not significant at BVPS-1. Hence, the issue addressed in Request for Additional Information (RAI)-2(b) is resolved.

In its response to RAI-2(c) in the supplement dated August 26, 2015, the licensee stated that the AMPs developed for both BVPS units follow the MRP-227-A guidance for managing the potential embrittlement and cracking degradation of the LSCs during the PEO. The BVPS-1 CASS LSC bodies are considered expansion components and are linked to the CRGT lower flange welds primary inspection component. Section 3.3.1 of MRP-227-A defines expansion component as one that is moderately susceptible to aging degradation. The expansion component is linked to a primary component that has a higher degree of susceptibility to experience the same aging degradation. Therefore, primary components are to be inspected first, and any detection of an active aging degradation in this component would require inspections on the corresponding expansion component. The BVPS-2 wrought (non-CASS) LSC bodies are considered expansion components and are linked to the upper core barrel flange weld primary inspection component.

The licensee provided two regulatory commitments in support of its responses to RAI-2(a) and RAI-2(c) in Attachment 2 of its letter dated August 26, 2015.

The first commitment is related to RAI-2(a), which includes a request to the licensee to provide a plant-specific analysis on the combined effects of IE and TE on LSCs at BVPS-1. This commitment is to update the BVPS-1 AMP for the CASS LSC bodies in accordance with the PWROG project resolution within 180 days of the completion of the project. The second commitment is related to RAI-2(c), in which the staff requested the licensee to clarify why the primary inspection links to a possible expansion inspection of the LSCs for both BVPS units are adequate. In this commitment, the licensee is to implement the revised MRP-227-A guidance for the BVPS-1 and BVPS-2 primary component links for the LSC bodies within 90 days of the revision to MRP-227-A.

Based on its review, the staff determined that plant-specific analysis on the combined effect of TE and IE at BVPS-1 is not necessary because LSCs at this unit are not susceptible TE. However, the plant-specific analysis should be focused on the effect of IE on LSCs at this unit. With respect to the commitment related to the selection of a suitable primary link for LSCs, the staff has since received MRP-227, Revision 1, for review and approval. The staff is currently reviewing this report, and it noted that a suitable primary link was included in the revised MRP-227 report. The staff's further evaluation of this issue is discussed in the following paragraph.

The NRC staff finds that reasonable control for the implementation of the above regulatory commitments is best provided through the licensee's administrative processes, including its AMPs as described in the updated final safety analysis report (UFSAR) and its commitment management program. The above regulatory commitments do not warrant the creation of regulatory requirements and are not relied upon to find the BVPS RVI AMPs acceptable.

Based on the staff's review, it concluded that even though TE is not an active mechanism in LSCs in the BVPS units, IE is still an active aging degradation mechanism. In Table 4-6 of MRP-227-A, LSCs were binned under the "Expansion" inspection category, and the corresponding primary link is the upper core barrel flange weld, which is a primary indicator of SCC, but not of IE. The staff noted that the primary indicator of the aging effect due to IE would be an RVI component (binned under the "Primary" inspection category) that is susceptible to aging degradation due to IE. Therefore, in order to identify aging effects due to IE, a more suitable primary link for LSCs must be selected. Any cracking due to IE would be identified in this primary link RVI component much sooner than in the LSCs. The licensee, in its response to RAI-2(c), stated that as part of its commitment, it will implement a primary link (lower core barrel girth weld) that is a better indicator of IE and is consistent with the guidance developed by the industry. Based on review and the closure of AI-2, the NRC staff concludes that the licensee adequately addressed AI-7. Therefore, the staff finds that the issue related to AI-7 is closed because (1) LSCs in BVPS-1 are not susceptible to TE and (2) the responses to RAI-2, which demonstrate that IE is satisfactorily addressed by the licensee.

3.3.7 Assessment of Resolution to AI-8

As addressed in Section 3.5.1 of the staff's SE for MRP-227-A, AI-8 requires that the licensee make a submittal for NRC review and approval to credit the implementation of MRP-227-A as part of an AMP for the RVI components at its facilities. The licensee complied with its commitment by submitting its AMP, which included I&E guidelines addressed in MRP-227-A. Therefore, the NRC staff concludes that the licensee adequately addressed AI-8.

3.4 Assessment of the Licensee's Responses to the TR Conditions in Section 4.1 of the SE for MRP-227A

With respect to the seven TR conditions in the staff's SE for MRP-227-A, the NRC staff reviewed Table 6-1, "Topical Report Condition Compliance to SE on MRP-227," in Enclosures A and B for BVPS-1 and BVPS-2, respectively, of the submittal. The NRC staff's evaluation is addressed in the following paragraphs.

TR Condition 1: Per Section 4.1.1, "High Consequence Components in the 'No Additional Measures' Inspection Category," of the staff's SE for MRP-227-A, the licensee added to Table C-2, "MRP-227-A Expansion Inspection and Monitoring Recommendations for Westinghouse-Designed Internals," of Enclosures A and B of the submittal, the upper core plate and lower support forging or casting, to its RVI inspection program. This addition is consistent with the guidelines in Table 4-6 of MRP-227-A and, therefore, the staff considers Condition 1 satisfied.

TR Condition 2: Per Section 4.1.2, "Inspection of Components Subject to Irradiation-Assisted Stress Corrosion Cracking," of the staff's SE for MRP-227-A, the licensee included the upper and lower core barrel cylinder girth welds and lower core barrel flange weld as Primary Components in Table C-1, "MRP-227-A Primary Inspection and Monitoring Recommendations for Westinghouse-Designed Internals," in Enclosures A and B of the submittal. Therefore, the staff considers Condition 2 satisfied.

TR Condition 3: Condition 3 is not applicable to Westinghouse-designed RVI components and, therefore, the staff did not evaluate this issue in this staff assessment.

TR Condition 4: A criterion for a minimum area of inspection coverage is addressed in this condition. This condition states that a minimum of 75 coverage coverage of the entire examination volume (i.e., including both accessible and inaccessible regions) of the RVI components and their welds, and a minimum sample size of 75 percent of the total population of like components (e.g., bolts) should be inspected. The licensee included this condition in Note 2 in Table C-2, "MRP-227-A Expansion Inspection and Monitoring Recommendations for Westinghouse-Designed Internals," of Enclosures A and B of the submittal. Therefore, the staff considers Condition 4 satisfied.

TR Condition 5: This condition states that a 10-year inspection frequency for baffle-former bolts in Westinghouse-designed reactors should be implemented following the initial or baseline required inspection. The licensee included this condition in Table C-1, "MRP-227-A Primary Inspection and Monitoring Recommendations for Westinghouse-Designed Internals," of Enclosures A and B of the submittal. Therefore, the staff considers Condition 5 satisfied.

TR Condition 6: This conditions states that subsequent re-examination for all "Expansion" inspection category components should be at a baseline 10-year re-examination interval. The licensee included this in Table C-2, "MRP-227-A Expansion Inspection and Monitoring

Recommendations for Westinghouse-Designed Internals," of Enclosures A and B of the submittal. Therefore, the staff considers Condition 6 satisfied.

TR Condition 7: Per Section 4.1.7, "Updating of MRP-227-A Appendix A," of the staff's SE for MRP-227-A, Appendix A of the MRP-227-A report was updated to include the operating experience related to the aging degradation of the RVI components in the PWR fleet, which references Section XI.M16A of the GALL report. As discussed in Section 3.2 of this staff assessment, the licensee evaluated operating experience related to RVI aging management. Therefore, the staff considers Condition 7 satisfied.

4.0 CONCLUSION

The NRC staff has reviewed the AMPs for the BVPS-1 and BVPS-2 RVI components submitted on January 27, 2014, as supplemented on August 26, 2015, and November 6, 2015. The NRC staff concludes that the BVPS-1 and BVPS-2 AMPs are acceptable, because they are consistent with the I&E guidelines of MRP-227-A, and the licensee addressed all eight applicant/licensee AIs specified in MRP-227-A appropriately.

The NRC staff's approval of the BVPS RVI AMPs does not reduce, alter, or otherwise affect current ASME Code, Section XI, Inservice Inspection (ISI) requirements, or any BVPS-1 or BVPS-2 specific licensing requirements related to ISI.

5.0 REFERENCES

1. Submittal letter from Eric A. Larson, FENOC, to the NRC, dated January 27, 2014 (Agencywide Documents Access and Management System (ADAMS) Package Accession No. ML14030A131).

2. Response to Request for Additional Information, dated August 26, 2015 (ADAMS Accession No. ML15239A710).

3. Response to Request for Additional Information, dated November 6, 2015 (ADAMS Accession No. ML15313A306).

4. WCAP-17789-NP, Revision 1, "PWR Vessel Internals Program Plan for Aging Management of Reactor Internals at Beaver Valley Power Station Unit 1" (ADAMS Accession No. ML14030A134).

5. WCAP-17790-NP, Revision 1, "PWR Vessel Internals Program Plan for Aging Management of Reactor Internals at Beaver Valley Power Station Unit 2" (ADAMS Accession No. ML14030A135).

6. "Materials Reliability Program: Pressurized Water Reactor Internals Inspection and Evaluation Guidelines (MRP-227-A)," December 2011 (ADAMS Accession No. ML12017A194).

7. NUREG-1929, "Safety Evaluation Report Related to the License Renewal of Beaver Valley Power Station, Units 1 and 2," Docket Nos. 50-334 and 50-412, FirstEnergy

Nuclear Operating Company, October 2009 (ADAMS Accession Nos. ML093020275 and ML093020276).

8. Submittal letter from EPRI, "Materials Reliability Program: Pressurized Water Reactor Internals Inspection and Evaluation Guidelines (MRP-227-Rev. 0)," dated January 12, 2009 (ADAMS Accession No. ML090160204).

9. Final NRC Safety Evaluation of MRP-227, Revision 1, "Materials Reliability Program: Pressurized Water Reactor Internals Inspection and Evaluation Guidelines (MRP-227, Revision 0)," dated December 16, 2011 (ADAMS Accession No. ML11308A770).

10. NUREG-1801, Revision 2, "Generic Aging Lessons Learned Report (GALL) – Final Report," December 2010 (ADAMS Accession No. ML103490041).

11. "Final License Renewal Interim Staff Guidance, LR-ISG-2011-04, Updated Aging Management Criteria for Reactor Vessel Internal Components for Pressurized Water Reactors" (ADAMS Accession No. ML12270A436).

12. NUREG-1800, Revision 2, "Standard Review Plan for Review of License Renewal Applications for Nuclear Power Plants – Final Report," December 2010 (ADAMS Accession No. ML103490036).

13. Submittal letter from PWROG of WCAP-17451-P, Revision 1, "Reactor Internals Guide Tube Wear – Westinghouse Domestic Fleet Operational Projections" (ADAMS Accession No. ML15041A106).

14. WCAP-17780-P, "Reactor Internals Aging Management MRP-227-A Applicability for Combustion Engineering and Westinghouse Electric Company Pressurized Water Reactor Designs," and MRP-227-A, Applicability Guidelines for CE and Westinghouse Pressurized Water Reactor Designs (ADAMS Accession No. ML13183A373) (non-public), submittal letter ADAMS Accession No. ML13183A372)).

15. MRP 2013-025, "MRP-227-A Applicability Template Guideline," dated October 14, 2013 (ADAMS Accession No. ML13322A454).

16. Office of Nuclear Reactor Regulation Evaluation of WCAP-17780-P, "Reactor Internals Aging Management MRP-227-A Applicability for Combustion Engineering and Westinghouse Electric Company Pressurized Water Reactor Designs," and MRP-227-A, Applicability Guidelines for CE and Westinghouse Pressurized Water Reactor Designs, August 2014 (ADAMS Accession No. ML14309A484).

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

18. "Materials Reliability Program: Aging Management Strategies for Westinghouse and Combustion Engineering PWR Internals (MRP-232), 1016593, Final Report,

December 2008" (ADAMS Accession Nos. ML091671780 (non-public), ML092250192 (non-public), and ML092230745 (non-public)).

19. Letter from Christopher I. Grimes, NRC, to Douglas J. Walters, NEI, "License Renewal Issue No. 98-0030, 'Thermal Aging Embrittlement of Cast Austenitic Stainless Steel Components'," dated May 19, 2000 (ADAMS Accession No. ML003717179).

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Date: October 7, 2016

The NRC staff's approval of the BVPS-1 and BVPS-2 RVI AMPs does not reduce, alter, or otherwise affect current American Society of Mechanical Engineers Code, Section XI, Inservice Inspection (ISI) requirements, or any BVPS-1 and BVPS-2 specific licensing requirements related to ISIs.

If you have any questions concerning this matter, please contact the Project Manager, Taylor Lamb, at (301) 415-7128 or Taylor.Lamb@nrc.gov.

Sincerely,

/RA REnnis for/

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Docket Nos. 50-334 and 50-412

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