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

March 30, 2015

Mr. Eric McCartney
Site Vice President
NextEra Energy Point Beach, LLC
6610 Nuclear Road
Two Rivers, WI 54241

SUBJECT: POINT BEACH NUCLEAR PLANT, UNITS 1 AND 2 – STAFF ASSESSMENT OF REACTOR VESSEL INTERNALS INSPECTION PLAN BASED ON MRP-227-A (TAC NOS. ME8235 AND ME8236)

Dear Mr. McCartney:

By letter dated December 19, 2011, as supplemented by letters dated August 16, 2012, March 15, 2013, April 18, 2014, and January 13, 2015, NextEra Energy Point Beach, LLC (NextEra) submitted an aging management program (AMP) for the reactor vessel internals (RVI) at Point Beach Nuclear Plant (PBNP), Units 1 and 2 for review. The PBNP AMP was developed based on the U.S. Nuclear Regulatory Commission (NRC) staff approved topical report MRP-227-A, "Material Reliability Program: Pressurized Water Reactor Internals Inspection and Evaluation Guidelines." The AMP was submitted to fulfill Regulatory Commitment 29 as documented in NUREG-1839, "Safety Evaluation Report Related to the License Renewal of the Point Beach Nuclear Plant, Units 1 and 2."

The NRC staff has completed its review of the PBNP reactor vessel internals AMP and concludes that it is acceptable because it is consistent with the inspection and evaluation guidelines of MRP-227-A, and NextEra has appropriately addressed all eight applicant/licensee action items specified in MRP-227-A. As such, Regulatory Commitment 29, as documented in Appendix A of NUREG-1839, is considered fulfilled. A copy of the staff's assessment is enclosed.

The NRC staff's approval of the PBNP reactor vessel internals AMP does not reduce, alter, or otherwise affect current American Society of Mechanical Engineers Boiler and Pressure Vessel Code, Section XI inservice inspection requirements, or any PBNP specific licensing requirements related to inservice inspection. The staff notes that Section 7.0, "Implementation Requirements," of MRP-227-A requires that the NRC be notified of any deviations from the "Needed" requirements.

If you have any questions concerning this matter, please contact the Project Manager, Mahesh Chawla at (301) 415-8371 or via e-mail at Mahesh.Chawla@nrc.gov.

Sincerely,

David L. Pelton, Chief
Plant Licensing Branch III-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. 50-266 and 50-301

Enclosure:
Staff Assessment

cc w/encl: Distribution via ListServ

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

STAFF ASSESSMENT BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AGING MANAGEMENT PROGRAM PLAN

FOR REACTOR VESSEL INTERNALS

NEXTERA ENERGY POINT BEACH, LLC

POINT BEACH NUCLEAR PLANT, UNITS 1 AND 2

DOCKET NOS. 50-266 AND 50-301

1.0 INTRODUCTION AND BACKGROUND

By letter dated December 19, 2011, as supplemented by letters dated August 16, 2012, March 15, 2013, April 18, 2014, and January 13, 2015 (Agencywide Documents Access and Management System (ADAMS) Accession Nos. ML113540301, ML12229A580, ML13077A356, ML14111A050, and ML15013A157, respectively), NextEra Energy Point Beach, LLC (the licensee) submitted an aging management program (AMP) for the reactor vessel internals (RVI) at Point Beach Nuclear Plant (PBNP), Units 1 and 2. The MRP-227-A report, "Pressurized Water Reactor (PWR) Internals Inspection and Evaluation Guidelines," and its supporting reports (ADAMS Accession Nos. ML12017A193 - ML12017A199) were used as the technical bases for developing PBNP's AMP. The licensee submitted the AMP with the intent to meet the schedule found in the letter to the U.S. Nuclear Regulatory Commission (NRC), dated April 23, 2010, for Commitment 29 from the License Renewal Safety Evaluation Report NUREG-1839, "Safety Evaluation Report Related to the License Renewal of the Point Beach Nuclear Plant, Units 1 and 2" (ADAMS Accession Nos. ML051190134 and ML053420137). The AMP included inspection and evaluation (I&E) guidelines for the RVI components at PBNP, Units 1 and 2.

2.0 REGULATORY EVALUATION

Title 10 of the *Code of Federal Regulations* (10 CFR) Part 54, "Requirements for the Renewal of Operating Licenses for Nuclear Power Plants," addresses the requirements for the renewal of nuclear power plant licenses. The regulation at 10 CFR Section 54.21 requires that each application for license renewal contain an integrated plant assessment (IPA) and an evaluation of time-limited aging analyses (TLAA). The plant-specific IPA shall identify and list those structures and components subject to an aging management review (AMR) and demonstrate that the effects of aging (cracking, loss of material, loss of fracture toughness, dimensional changes, loss of preload) will be adequately managed so that their intended functions will be maintained consistent with the current licensing basis for the period of extended operation (PEO) as required by 10 CFR 54.29(a). In addition, 10 CFR 54.22 requires that a license renewal application (LRA) include any technical specification changes or additions necessary to manage the effects of aging during the PEO as part of the LRA.

Enclosure

Structures and components subject to an AMR shall encompass those structures and components that (1) perform an intended function, as described in 10 CFR 54.4, without moving parts or without a change in configuration or properties and (2) are not subject to replacement based on a qualified life or specified time period. These structures and components are referred to as "passive" and "long-lived" structures and components, respectively.

On January 12, 2009, the Electric Power Research Institute (EPRI) submitted for NRC staff review and approval, the Materials Reliability Program (MRP) Report MRP-227, Revision 0, "PWR Internals Inspection and Evaluation Guidelines" (ADAMS Accession No. ML090160204), which was intended as guidance for applicants in developing their plant-specific AMP for RVI components. Revision 1 to the final safety evaluation (SE) regarding MRP-227, Revision 0, was issued on December 16, 2011 (ADAMS Accession No. ML11308A770). The SE contains specific conditions on the use of the topical report and applicant/licensee action items (A/LAI) that must be addressed by those utilizing the topical report as the basis for a submittal to the NRC. On January 9, 2012, EPRI published the NRC-approved version of the topical report, 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 vessels and also provides inspection and evaluation guidelines for PWR applicants to use in their plant-specific AMPs. MRP-227-A provides the basis for renewed license holders to develop plant-specific inspection plans to manage the effects of aging on RVI components, as described by their license renewal commitment.

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 (ASME) Boiler and Pressure Vessel 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 non-long-lived components such as fuel assemblies and reactivity control assemblies, or active components such as nuclear instrumentation because these components are not typically within the scope of the components that are required to be subject to an AMR, as defined by the criteria set in 10 CFR 54.21(a)(1).

Subsequent to the submittal of MRP-227 and prior to the issuance of the SE on MRP-227, NUREG-1800, Revision 2, "Generic Aging Lessons Learned (GALL) Report – Final Report" (the GALL Report, Revision 2, ADAMS Accession No. ML103490041), 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 Report, Revision 2 was published prior to the issuance of the final SE of MRP-227-A, the staff published 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), which modifies the guidance of AMP XI.M16A to be consistent with MRP-227-A.

3.0 TECHNICAL EVALUATION

The staff's review of the PBNP RVI AMP focused on determining whether the licensee met its license renewal commitment to incorporate the applicable recommendations for augmented inspections and techniques resulting from the industry effort on RVI (summarized in

MRP-227-A) into the plant-specific AMP. Therefore, the most important aspects of the staff's review are the review of the augmented inspections specified in the RVI AMP, and the licensee's responses to the A/LAIs from Revision 1 to the final SE regarding MRP-227, Revision 0. Other than as noted in the following sections, the information included in the AMP did not require a specific review by the staff.

3.1 Organization of the PBNP RVI AMP

Sections 1.0, "Purpose," 2.0, "Background," 3.0, "Responsibilities," 5.0 "References," and Attachment A, "Figures," of the AMP have no specific technical information which would affect the review and approval of the PBNP units' inspection plan. Therefore, the focus of the NRC staff's evaluation is related to Section 4.0, "Procedure," which describes how the program works as well as the expansion criteria, evaluation, repair and replacement strategy, implementation schedule, and commitment tracking process. There is considerable discussion in Section 4.3.14 of the unique plant-specific operating experience with RVI components. The submittal ends with Attachments B through D, which summarize the augmented primary, expansion, and existing inspections incorporated into the PBNP inservice inspection (ISI) plan for the license renewal period as well as Attachment E, which contains the examination acceptance and expansion criteria.

3.2 Evaluation of Section 4.0 of PBNP RVI AMP

The NRC staff's review focused on existing, plant-specific programs and operating experience (OE). The existing programs, summarized in Section 4.2, are used for aging management of the reactor coolant system as well as RVI at PBNP. They include the ASME Section XI Subsections IWB, IWC, and IWD Inservice Inspection Program for core support structures, augmented inspections on remaining RVI components addressed in MRP-227, and the Thimble Tube Inspection Program, which is based on AMP XI.M37, "Flux Thimble Tube Inspection," in NUREG-1801, Revision 1 "Generic Aging Lessons Learned (GALL) Report," commonly referred to as the GALL Report.

In reviewing this section, the NRC staff noted that these programs are also included in Attachment D, but there is only limited discussion of the extent of aging degradation that has been detected in the programs so far. To clarify the extent of aging, the staff issued several requests for additional information ((RAIs) 3, 4, 6, and 7) by emails dated June 7 and July 10, 2012 (ADAMS Accession Nos. ML12159A113 and ML12198A050, respectively), that requested the licensee to summarize the existing program results to date and plans for the future.

In its August 16, 2012, response to RAI-7, the licensee noted that the ASME Code Section XI inspections at PBNP have not discovered any signs of significant aging degradation. Given the industry experience with the cracking of control rod guide tube (CRGT) support pins (commonly referred to as split pins) fabricated from Alloy X-750, the split pins at PBNP were replaced with Type 316 stainless steel because Type 316 is more resistant to stress corrosion cracking (SCC) than Alloy X-750. The licensee does not consider the split pins as part of the core support structure in their ASME Section XI program, but the problem documented in MRP-227-A, Appendix A was noted during the ASME Code Section XI inspections. The licensee noted that future inspections will look at the accessible portions of the split pins whenever the core barrel is removed, but the split pins are not officially part of the required ASME Section XI program.

In its August 16, 2012, response to RAI-4, the licensee noted that no cracking has been found in the clevis insert assembly from either unit during the required ASME Section XI program inspections. The bolts, which are part of the assembly, were fabricated from Alloy X-750, similar to the split pins. In addition, there is no plant-specific inspection history related to Alloy A-286, Type 347 (excluding baffle-former bolts), precipitation hardened stainless steel, or Type 431 stainless steel because they are not used in the RVI components at PBNP.

It its August 16, 2012, response to RAI-3 and RAI-6 regarding thimble tubes, the licensee summarized the experience to date. The original tubes in both units were replaced in 1985 due to internal blockages. No leakage was evident. The replacement tubes were fabricated from Type 316 stainless steel with a slightly larger inside diameter to prevent blockage. For Unit 1, five tubes were replaced a second time in 1998. As part of the current PBNP Thimble Tube Condition Assessment Program, eddy current examinations are performed on a regular basis that is determined by the previous inspection results, but not more than 6 years between inspections. The licensee is in the process of upgrading the inspection program to align with the GALL Report, Revision 2, AMP XI.M37.

The NRC staff has reviewed the RAI responses regarding the existing programs from the August 16, 2012, letter along with Section 4.2 of the December 19, 2011, submittal, MRP-227-A and the LR-ISG guidance. The staff finds that the ASME Section XI inspections have been effective in detecting the cracking of split pins even though the split pins are not required to be inspected under the ASME Code Section XI program. For the clevis insert assembly, similar cracking of the Alloy X-750 bolts along with wear in the assembly have been noted in another domestic PWR as part of the ASME Section XI ISI, but its absence in the PBNP units cannot be assumed for the PEO. An RAI response related to the staff's review of the Indian Point Nuclear Generating Units 2 and 3 (IP2 and IP3) RVI Inspection Plan, which is a plant of similar design to PBNP (ADAMS Accession No. ML14310A803), provided a technical justification for the adequacy of the ASME Section XI VT-3 inspections for the clevis insert bolts. This justification was based on the design functions of lower radial support system (LRSS) being maintained even if all clevis insert bolts in a clevis insert were to fail, the high degree of redundancy in the system, and the capability of VT-3 inspection to detect evidence of degradation of the bolts or the clevis inserts. Further, in a June 2014 public meeting, the Pressurized Water Reactor Owner's Group (PWROG) provided an update on its generic evaluation of clevis insert bolt functionality, which concluded that all clevis insert design configurations are inherently safe, loose parts are captured (with the possible exception of lock bars), and there is no single point failure leading to loss of function (ADAMS Accession No. ML14163A520). The PWROG further concluded that concerns related to the clevis insert bolts are primarily commercial in nature, and that visual inspection of wear surfaces and general condition will provide the appropriate level of aging management without the need for bolt inspections. The staff reviewed the information from IP2 and IP3 and the PWROG, and concluded that the same conclusions are applicable to PBNP, Units 1 and 2 based on the similar design of the RVI and the LRSS. Therefore, the staff concludes that continued inspection as part of the ASME Section XI program with the 10-year interval for PBNP is appropriate and consistent with the prior approval for a different plant.

In addition, the NRC staff notes that the licensee has begun the process of updating its flux Thimble Tube Inspection Program to the latest NRC-approved guidance. Therefore, based on the NRC staff's review discussed above, the issues related to the existing inspection programs at PBNP (RAIs 3, 4, 6, and 7) are resolved, and the staff finds that the existing inspection programs, summarized in Attachment D, are acceptable.

The plant-specific OE is summarized in Section 4.3.14 of the December 19, 2011, submittal. Two significant pieces of OE for PBNP are discussed in this SE because they had significant input into the industry's planning for MRP-227: (1) PBNP participated in a PWR owners group program that involved a one-time, augmented ultrasonic testing (UT) inspection of baffle-former (B/F) bolts under which PBNP Unit 2 B/F bolts were inspected, and (2) PBNP participated in a separate PWR owners group program looking at CRGT guide card assemblies.

The B/F inspection program at PBNP Unit 2 ran from December 1998 to January 1999, and was the domestic industry's response to a cracking problem noted at several foreign PWRs. A number of bolts were found in PBNP Unit 2 with crack-like indications. The number of bolts required to guarantee the structural margins of the B/F joints were replaced, including all bolts with UT indications. The replacement bolts were made from a material that is considered to be more resistant to irradiation-assisted SCC (IASCC).

In RAI-2, sent by email dated June 7, 2012 (ADAMS Accession No. ML12159A113), the NRC staff requested more specific information related to the B/F bolt inspection and replacement program and the future MRP-227-A inspections. Given the information related to the irradiation conditions of the B/F bolts already published in MRP-51, "Materials Reliability Program-Hot Cell Testing of Baffle/Former Bolts Removed from Two Lead PWR Plants" (ADAMS Accession No. ML053180527), the staff asked for estimated irradiation conditions at the time of the first MRP-227-A inspection for the bolts that have been in service since the beginning. Based on the measured fluence values reported in MRP-51, the NRC staff was concerned that the irradiation conditions at the time of the first inspections may not be representative of the assumptions made in the generic screening for MRP-227.

In its letter dated August 16, 2012, the licensee provided a summary from EPRI Technical Report TR-114779, "Inspection and Replacement of Baffle to Former Bolts at Point Beach-2 and Ginna Processes, Equipment Design, and Equipment Qualification," of the inspection and replacement processes employed at PBNP and planned for future MRP-227-A inspections. First, all of the 728 B/F bolts in Unit 2 were inspected, and only 14 bolts were verified as containing substantial cracks. The most likely cause of cracking appeared to be a combination of irradiation-assisted SCC (IASCC) and fatigue; the bolts with indications did not conform to any pattern related to the amount of neutron exposure they had received.

The licensee noted that the replacement B/F bolts were made from strain-hardened Type 316 stainless steel that has been screened with additional controls and testing to ensure optimum material properties and resistance to cracking. The design, fabrication, and installation of the replacement bolts have been modified to enhance the service performance and minimize problems related to inspection of the components.

Finally, the August 16, 2012, letter notes that all 728 B/F bolts are included in the MRP-227-A "Primary" inspections for each unit; the inspection method will be the same, but improvements have been made to the UT equipment to address issues discovered during the first exams. By the time the first MRP-227-A inspections on the B/F bolts were done in 2013 (Unit 1) and will be done in 2015 (Unit 2), the operating time will be about 34 effective full power years (EFPY) for

Unit 1 and 36 EFPY for Unit 2; this translates into a neutron exposure of 73.2 displacements per atom (dpa) for Unit 1 and 77.7 dpa for Unit 2.

The NRC staff has reviewed the RAI responses regarding the OE from the August 16, 2012, letter, along with Section 4.3.14 of the December 19, 2011, submittal, NUREG-1839 (SER for 2005 LRA), SER for 2009 extended power uprate (EPU), MRP-51, and MRP-227-A, Appendix A. The plant-specific OE is consistent with the generic OE found in MRP-227-A and the recent discovery of cracked B/F bolts in a domestic PWR. The lack of a clear pattern of cracking that is related to the neutron exposure indicates that neutron exposure is not the only factor that affects the degradation and other factors are significant contributors. Given the mixture of original and replacement bolts and the range of neutron exposure on each bolt, the MRP-227-A UT inspection of each B/F bolt in the two units should provide a clear picture of any on-going degradation of the B/F bolts. Based on the NRC staff's review of the history and inspection plans for B/F bolts at PBNP, the concern expressed by the staff related to RAI-2 is resolved, and the staff finds that the planned primary inspections, included in Attachment B, are acceptable.

The second piece of significant OE was related to an owners group inspection program to inspect CRGT guide cards at Unit 1, which took place during the 2008 outage. The inspection results are being evaluated to ensure compliance with MRP-227 specifications.

The NRC staff noted that MRP-227-A provides only general descriptions of the examination coverage for several primary components, including the CRGT guide cards. For example, the licensee is required to inspect only 20 percent of the CRGT guide card assemblies according to Attachment B of the submittal. The staff was concerned that for components for which 100 percent coverage is not required, or 100 percent of the component population is not accessible for inspection, the areas or components being inspected may be less susceptible to degradation than those areas or components that are not being inspected. Therefore, in RAI-5 (ADAMS Accession No. ML12159A113), the NRC staff asked for more specifics on what will be considered in the selection process for the primary components listed in Attachment B, and specifically that the licensee explain if the planned MRP-227-A Primary inspections will cover the most susceptible areas determined from plant-specific OE and/or subject to higher applied stress, and to note any limitations due to accessibility so that any potential degradation can be identified.

In its letter dated August 16, 2012, the licensee provided a detailed response to RAI-5. Some wear was noted during the 2008 inspections done for Unit 1. For the guide card assemblies, 100 percent of the assemblies for both units will receive the prescribed visual testing (VT)-3 examination, which is based on the recommendations contained in WCAP-17020-P, "Point Beach Unit 1 Upper Internal Guide Tube -- Guide Card Wear Evaluation."

For the CRGT lower flange welds, all of the individual assemblies on the periphery will be inspected, but only the outer weld surfaces and adjacent base metal are accessible for the prescribed enhanced visual testing (EVT)-1 inspections. The licensee noted that the regions to be inspected have similar applied stresses as the welds on the interior of the assemblies; and the minimal gap between assemblies will not allow access for 100 percent coverage during the EVT-1 inspections. Furthermore, no degradation has been found during previous VT-3 inspections of the CRGT lower flange welds.

Finally, for the core barrel assembly welds, the licensee plans to follow the MRP-227-A guidance, which is inspection of 100 percent of one side of the accessible surfaces around the welds. The licensee pointed out that the stresses should be the same on either side of the welds and some of the welds are only accessible when the core barrel is pulled. Furthermore, the lower core barrel cylinder girth weld has limited access due to the surrounding components. Plant-specific OE with VT-3 examinations of the core barrel have not found any degradation in the past.

The NRC staff has reviewed the response to RAI-5 regarding the VT-3 inspection of the Primary components. In each case, the licensee has provided specific details related to the coverage expected with an explanation for cases where reduced coverage is anticipated, and the areas or components to be inspected are equally or more susceptible to degradation than those not being inspected. The staff notes that PBNP is planning to go beyond the minimum MRP-227-A required coverage (i.e., 20 percent of assemblies) to inspect all guide card assemblies for both units based on plant-specific OE that has come from a proactive owners group inspection program for Unit 1. Inspections for the other Primary components, such as the lower flange welds and the core barrel assembly welds, do follow the MRP-227-A generic plan even with plant-specific limitations anticipated due to restricted access for EVT-1 examinations. Based on the RAI response related to the planned inspections, which meet the minimum requirements for coverage and the expanded augmented inspections for the Primary components, the NRC staff has determined that its concern expressed in RAI-5 related to how potential degradation can be identified is resolved.

3.3 Demonstration of Compliance with MRP-227-A Applicant/Licensee Action Items

3.3.1 A/LAI 1 – Plant-Specific Applicability of MRP-227-A to PBNP

Section 4.2.1 of the SE for MRP-227-A states in part that

> 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] FMECA and functionality analyses for reactors of their design (i.e., Westinghouse, CE, or B&W) which support MRP-227 and describe the process used for determining plant-specific differences in the design of their RVI components or plant operating conditions, which result in different component inspection categories. The applicant/licensee shall submit this evaluation for NRC review and approval as part of its application to implement the approved version of MRP-227.

To resolve the generic issue of the information needed from the licensee to resolve A/LAI 1, the staff asked a series of questions in RAI-1 to help verify that the values of fluence, temperature, stress, and material documented in MRP-191, "Materials Reliability Program: Screening, Categorization, and Ranking of Reactor Internals Components for Westinghouse and Combustion Engineering PWR Design," and MRP-232, "Materials Reliability Program: Aging Management Strategies for Westinghouse and Combustion Engineering PWR Internals" were bounding for PBNP. In its MRP-191 letter dated August 16, 2012, the licensee provided a listing for specific components with the value of fluence, temperature, stress, and material for a typical plant (taken from MRP-191) vs. PBNP. In each case, the PBNP value was bounded by that of a typical plant. Therefore, the licensee stated that there was no plant-specific reason to change the MRP-227-A inspection requirements.

The NRC staff reviewed the response for A/LAI 1 in the August 16, 2012 letter and noted that the licensee did not mention that an EPU for each unit was approved by the NRC on May 5, 2011. The MRP has determined that the MRP-227-A inspection plan should be applicable to all domestic PWR units as of May 2007. In discussions with the MRP, the MRP indicated that any plant with an EPU after 2007 was not considered during the development of the inspection plan. The staff has reviewed the licensee's 2009 submittal for the EPU (ADAMS Accession No. ML091250566) and noted two examples where the PBNP stress or fluence value after the EPU was not bounded by the typical plant listed in the response to RAI-1.

Given the uncertainty related to the effects of the EPU at PBNP on the MRP-227-A inspection guidelines, the NRC staff issued RAI-1a, dated January 31, 2013 (ADAMS Accession No. ML13036A300), asking the licensee to review its current licensing basis, including the EPU, and list all components that are not bounded by the assumptions on stress, temperature, and fluence used in the development of MRP-227, and to evaluate the need to change any of the inspection guidelines for any component that is not bounded by the typical plant parameters that were used in the development of MRP-227.

In its letter dated April 18, 2014, the licensee provided a response to RAI-1a. The licensee went back to a scoping study for the EPU that was done to assess any potential adverse impact of the EPU on the plant-specific implementation of MRP-227-A. The study considered both design and operational characteristics of the two PBNP units. In all cases under the EPU conditions, the plant-specific parameters were bounded by the original design basis or were shown to be acceptable for the PEO. The study concluded that the EPU had no impact, adverse or otherwise, on the plant-specific implementation of MRP-227 guidelines. A second review done in response to RAI-1a was initiated to look at how the neutron fluence, temperature, materials, and stress values for PBNP with the EPU conditions compare to those used in the development of MRP-227-A. The conclusions from the second study confirmed that from the initial scoping study that was done for the EPU, both units at PBNP meet the requirements for application of MRP-227-A.

Regarding the two examples cited by the NRC staff in the request for RAI-1a where the stress or fluence with the EPU was not bounded by the typical plant, the licensee noted that there were cases where the stresses for the most severe, normal and upset conditions did exceed those reported for the typical plant, but that the normal operational stresses always would be bounded for the subject component in MRP-191. For fluence, the licensee pointed out that the B/F assembly is located in Region 6, per MRP-191, Section 4.3.2, where the fluence is predicted to be > 5×10^{22} n/cm^2, E > 1 MeV (equal to > 75 dpa) with no upper limit, the maximum fluence from Section 2.1.4.2.2 of the 2009 EPU submittal is consistent with the definition for Region 6 in MRP-191.

The NRC staff reviewed the licensee's response to RAI-1a and finds that the description of the process used to answer the request is comprehensive and has clarified the two examples noted in the request. Specifically, the original response to RAI-1, item iii for the barrel-former bolts in the August 16, 2012, letter, listed the neutron fluence value as 5×10^{22} n/cm^2, E > 1 MeV, for both the typical plant and PBNP. Because the component is found in Region 6, as defined in MRP-191, the value should have been listed as > 5×10^{22} n/cm^2, E > 1 MeV. The staff notes that this response to RAI-1a is adequate and resolves the specific issues raised in the RAI.

To resolve the generic issue of the information needed from licensees to resolve A/LAI 1, a series of public and non-public meetings were conducted, at which the NRC, Westinghouse, the 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, "Reactor Internals Aging Management MRP-227-A Applicability for Combustion Engineering and Westinghouse Pressurized Water Reactor Designs," and it 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. The staff documented its review of WCAP-17780-P in an evaluation dated November 7, 2014 (ADAMS Accession No. ML14309A484).

As a result of the technical discussions with the NRC staff, a technical basis was developed for the response to A/LAI 1 (ADAMS Accession Nos. ML13042A048 and ML13067A262, respectively). For a licensee to demonstrate that its plant is bounded by the MRP-227-A evaluation for originally licensed and uprated conditions, it would need to provide a satisfactory response to two questions by each Westinghouse and CE unit. A third question is applicable to those plants that have implemented an EPU after 2007.

Question 1: Does the plant have non-weld or bolting austenitic stainless steel (SS) components with > 20 percent cold work, and, if so, do the affected components have operating stresses greater than 30 ksi? (If both conditions are true, additional components may need to be screened in for 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?

Question 3: Are the peak internal metal temperatures within the assumptions made in developing MRP-227-A?

The issues raised by Question 1 have already been addressed by the response to RAI-1a in the April 18, 2014, letter with a direct comparison between the PBNP RVI components and those contained in MRP-191, Table 4-4. The review did not reveal any non-weld or bolting austenitic stainless steel components with ≥ 20 percent cold work.

In an email dated November 21, 2014, the NRC staff asked a follow-up question to RAI-1a, requesting the licensee provide answers to Questions 2 and 3 above.

In response to Question 2, by letter dated January 13, 2015 (ADAMS Accession No. ML15013A157), the licensee stated that both PBNP Units 1 and 2 comply with the MRP-227-A Applicability Guidelines, which are quantitative criteria to allow a licensee to assess whether a particular plant has atypical fuel design or fuel management. For a Westinghouse design plant such as PBNP, these criteria and the PBNP responses are as follows:

(1) The heat generation rate must be ≤ 68 Watts/cm^3. At PBNP, the highest heat generation figure of merit is 63.6 Watts/cm^3; the plant-specific value is bounded by the MRP guidance threshold value.

(2) The maximum average core power density must be less than 124 Watts/cm^3. At PBNP, the highest maximum average core power density is 103 Watts/cm^3; the plant-specific value is bounded by the MRP guidance threshold value.

(3) The active fuel to upper core plate (UCP) distance must be > 12.2 inches. At PBNP, the distance between the active fuel and the UPC is 13.2 inches; the plant-specific value is bounded by the MRP guidance threshold value.

In response to Question 3, by letter dated January 13, 2015, the licensee identified the uprated plant condition and restated the responses used for Questions 1 and 2 above, following the guidance listed in MRP 2013-025, "MRP-227-A Applicability Guidelines for Combustion Engineering and Westinghouse Pressurized Water Reactor Designs, Enclosure to MRP Letter 2013-025, October 14, 2013" (ADAMS Accession No. ML13322A454). The NRC staff notes that MRP-51 includes plant-specific metal temperatures for the cold-worked Type 316 B/F bolts removed in January 1999 from Unit 2 for the period when a low-leakage core was employed. The peak temperature for the low-leakage core period, which will be representative of the PEO, is only slightly above the nominal temperature for the B/F bolts in MRP-191 (> 608 °F). For the remainder of the PEO under the EPU conditions stated in the licensee's response to Question 3, the staff considers the heat generation figure of merit to be an operational factor that will influence peak metal temperature, a higher heat generation figure of merit should translate into higher peak internal metal temperatures. Because the highest calculated heat generation figure of merit for PBNP is 63.6 Watts/cm^3, which is below the threshold value of 68 Watts/cm^3, the staff considers that PBNP is operating in a manner consistent with the assumptions made in developing MRP-227-A.

Based on the submitted responses to the follow-up to RAI-1a, the NRC staff concludes that the licensee has followed the guidelines related to the fuel management issue addressed in MRP-227-A Applicability Guidelines. Implementation of I&E guidelines addressed in MRP-227-A would be valid during the PEO, and the issues raised in RAI-1 and -1a are resolved.

The licensee adequately addressed the three questions for which the NRC staff determined additional plant-specific information was necessary to verify applicability of MRP-227-A to PBNP–(1) cold work induced stress; (2) fuel management; and (3) peak internal metal temperature--by confirming that PBNP complies with the criteria defined in the MRP-227-A Applicability Guidelines. Furthermore, the licensee confirmed that PBNP will continue to comply with these limits during the PEO. Therefore, the staff finds that the licensee has addressed concerns related to A/LAI 1 for PBNP.

3.3.2 A/LAI 2 – RVI Components within Scope of License Renewal

Action Item 2 in the NRC staff's SE for MRP-227-A requires that an applicant/licensee be responsible for identifying which RVI components are within the scope of license renewal for their facility. As part of its April 18, 2014, response to RAI-1a, the licensee noted that it performed the scoping and screening of the RVI components per the requirements of the license renewal process. The RVI materials used at the PBNP units are largely consistent with the materials specified in MRP-191, which was used as a technical basis document for the development of I&E guidelines that are addressed in MRP-227-A for CE and Westinghouse units. There were two categories of inconsistency related to the material type, either (1) the plant documentation identified the subject component as either wrought austenitic stainless steel or cast austenitic stainless steel (CASS) and MRP-191 lists the material as wrought austenitic stainless steel, or (2) a different wrought stainless steel is used. For the two units, the components in question were located in the upper internals assembly (guide cards/plates, housing plates, brackets, clamps, terminal blocks, and conduit straps) and lower internals assembly (bottom mounted instrument column cruciforms and secondary core support guide posts). The inconsistencies were reconsidered by a FMECA expert panel. Based on the generic FMECA group number assigned to the component during the original screening, the likelihood of failure, and potential subsequent damage due to failure, the panel agreed that the inconsistencies in materials used in the RVI for PBNP did not require any additional action to conform to the MRP-227-A guidelines.

The NRC staff has reviewed the licensee's response to RAI-1a and finds that the process used by the licensee to evaluate the inconsistencies in the PBNP components is similar to that used in the generic assessment done for MRP-232. The fact that some subject components were fabricated from CASS instead of wrought austenitic stainless steel would mean that they would screen in for thermal embrittlement, but because the components are not safety significant and do not receive a significant neutron dose, the generic FMECA ranking would not change. In a similar fashion, the use of a different wrought austenitic stainless steel would not change the screening process for aging mechanisms and the generic FMECA ranking would not change. Therefore, because all of the inconsistencies in the PBNP components, compared with that used in MRP-191, were considered by the process described in MRP-232, the staff finds that the licensee has adequately addressed A/LAI 2.

3.3.3 A/LAI 3 – Evaluation of Plant-Specific Existing Programs

Action Item 3 in the NRC staff's SE for MRP-227-A requires an applicant/licensee to perform a plant-specific evaluation of its existing programs; for Westinghouse-designed units like PBNP, this refers to the split pins. From Section 4.2 of the December 19, 2011, submittal, the licensee described how it has replaced the original Alloy X-750 split pins at both units with ones manufactured from cold-worked Type 316 austenitic stainless steel. In its August 16, 2012, response to RAI-7, the licensee outlined its plan going forward to perform visual inspections of the accessible portions of the split pins whenever the core barrel is removed. Based on the licensee's replacement of the original Alloy X-750 split pins and its response to RAI-7, the staff considers that the licensee has adequately addressed A/LAI 3.

3.3.4 A/LAI 4 – Babcock &Wilcox Units, Upper Flange Stress Relief

Action Item 4 of the NRC staff's SE for MRP-227-A is applicable to the RVI components designed by Babcock & Wilcox (B&W), and therefore, is not applicable to the PBNP units.

3.3.5 A/LAI 5 – Application of Physical Measurements

Action Item 5 requires that an applicant/licensee for a Westinghouse unit identify the plant-specific acceptance criteria to be applied when performing physical measurements for loss of compressibility due to loss of load for the hold-down springs. Loss of load is applicable to more a susceptible material like Type 304 stainless steel. Type 403 stainless steel has a higher yield strength and is not considered susceptible. As part of the MRP-227-A review, the NRC staff determined that physical measurements for loss of compressibility for Type 403 stainless steel hold-down springs are not necessary. Since Type 403 springs are used at PBNP units, no physical measurements of the hold-down spring are required. As part of the OE included in Section 4.2 of the submittal, the licensee did report that Westinghouse measured the height of the core barrel hold-down springs for Unit 2 in 2006 and the result confirmed that the hold-down capacity was adequate for the PEO. Given that Type 403 stainless steel is used for the hold-down springs, the staff concludes that A/LAI 5 is not applicable to the PBNP units.

3.3.6 A/LAI 6 – B&W Units, Evaluation of Inaccessible Components

Action Item 6 of the NRC staff's SE for MRP-227-A is applicable to the RVI components designed by B&W, and therefore, is not applicable to the PBNP units.

3.3.7 A/LAI 7 – Plant-Specific Evaluation of Components Fabricated from CASS

Action Item 7 requires an applicant/licensee to perform a plant-specific analysis demonstrating that the MRP-227 recommended inspections will ensure functionality of the set of components made of CASS, martensitic stainless steel, or precipitation hardened stainless steel during the PEO. For Westinghouse-design RVI, A/LAI 7 specifically identifies the lower core support columns as a component that may be CASS that should be addressed by the response to the A/LAI. The action item was not explicitly discussed in the initial December 19, 2011, submittal, so the NRC staff asked RAI-4 and RAI-8 to document how the licensee can address A/LAI 7. In its August 16, 2012, response to RAI-4, the licensee noted that there are no RVI components fabricated from martensitic stainless steel or precipitation hardened stainless steel. In its March 15, 2013, letter, the licensee responded to RAI-8 by listing all of the components fabricated from CASS, but deferred the plant-specific analysis until the work to answer A/LAI 2 was completed. In the April 18, 2014, responses to RAI-8 and RAI-1a, the licensee concluded that, based on the FMECA process, because there was a low likelihood of failure and minimal damage would be expected in the unlikely event of a failure of a component fabricated from CASS, no change to the MRP-227-A guidelines is needed for implementation at PBNP. In addition, the licensee noted that the lower support columns at the PBNP units are made of forged (non-cast) stainless steel and, therefore, the lower support columns are not covered by A/LAI 7. Therefore, based on the fact that the lower support columns for the PBNP units are not fabricated from CASS and the FMECA results for other plant-specific CASS components discussed in the licensee's responses to RAI-4, RAI-8 and RAI-1a, the NRC staff concludes that the licensee has adequately addressed A/LAI 7.

3.3.8 A/LAI 8 – Submittal of RVI AMP for Staff Review and Approval

Action Item 8 of the NRC staff's SE for MRP-227-A requires that the applicant/licensee submit an AMP for the RVI components that is consistent with I&E guidelines addressed in MRP-227-A. The licensee's AMP program elements were included in Section 4 of the December 19, 2011 submittal and its evaluation was addressed in Section 3.2 of this SE. Therefore, the staff concludes that the licensee has adequately addressed A/LAI 8.

3.4 Conditions in the Staff's SE for the MRP-227

With respect to the seven conditions in the NRC staff's SE for MRP-227, the staff reviewed the submittal dated December 19, 2011, as supplemented by letters dated August 16, 2012, March 15, 2013, April 18, 2014, and January 13, 2015, and concludes that the licensee, in its AMP for the RVI components, included all the conditions that are addressed in the staff's SE for MRP-227. Therefore, the NRC staff accepts the licensee's inspection program for the RVI components.

4.0 CONCLUSION

The NRC staff has reviewed the inspection plan for the PBNP, Units 1 and 2 RVI components and concludes that the PBNP inspection plan is acceptable because it is consistent with the I&E guidelines of MRP-227-A, and the licensee has addressed all of the applicable applicant/licensee action items specified in MRP-227-A appropriately. Consequently, the licensee meets the license renewal commitment 29 listed in Appendix A to NUREG-1839 for the RVI components at PBNP.

The NRC staff's approval of the PBNP RVI AMP does not reduce, alter, or otherwise affect current ASME Code, Section XI ISI requirements, or any PBNP specific licensing requirements related to ISI. The licensee must follow the implementation requirements as defined in Section 7.0 of MRP-227-A, which require that the NRC be notified of any deviations from the "Needed" requirements.

Principal Contributor: P. Purtscher, NRR
Date: March 30, 2015

If you have any questions concerning this matter, please contact the Project Manager, Mahesh Chawla at (301) 415-8371 or via e-mail at Mahesh.Chawla@nrc.gov.

Sincerely,

/RA/

David L. Pelton, Chief
Plant Licensing Branch III-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. 50-266 and 50-301

Enclosure:
Staff Assessment

cc w/encl: Distribution via ListServ

DISTRIBUTION:
NON-PUBLIC
LPL3-1 r/f
RidsNrrDorlLpl3-1Resource
RidsNrrPMPointBeach Resource
RidsOgcRp Resource
RidsNrrDeEvib Resource
PPurtscher, NRR
RidsAcrsAcnw_MailCTR Resource
RidsNrrDorlDpr Resource
RidsRgn3MailCenter Resource
RidsNrrLAMHenderson Resource

ADAMS Accession No.: ML15079A087 * via memorandum

OFFICE	LPL3-1/PM	LPL3-1/LA	DE/EVIB	DLR/RARB	LPL3-1/BC
NAME	MChawla	MHenderson	SRosenberg*	DMorey (JMedoff for)	DPelton
DATE	03/27/2015	03/25/2015	03/18/2015	03/27/2015	03/30/2015

OFFICIAL RECORD COPY