NEXTera ENERGY POINT BEACH

August 16, 2012

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U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, DC 20555

Point Beach Nuclear Plant, Units 1 and 2
Dockets 50-266 and 50-301
Renewed License Nos. DPR-24 and DPR-27

Reactor Vessel Internals Inspection Plan
Response to Request for Additional Information

References:
(1) NextEra Energy Point Beach, LLC letter to NRC, dated December 19, 2011, License Renewal Commitment, Reactor Vessel Internals Program Submittal (ML113540301)
(2) NRC electronic mail to NextEra Energy Point Beach, LLC, dated June 7, 2012, Point Beach Units 1 and 2 – Draft RAI on the Reactor Vessel Internals Inspection Plan (TAC ME8235 and ME8236) (ML12159A113)
(3) NRC electronic mail to NextEra Energy Point Beach, LLC, dated July 10, 2012, Point Beach Units 1 and 2 – Draft Request for Additional Information re: Reactor Vessel Internals Inspection Plan (TAC Nos. ME8235 and ME8235) (ML12198A050)

NextEra Energy Point Beach, LLC (NextEra) submitted the Point Beach Nuclear Plant (PBNP) program NP 7.7.30, Reactor Vessel Internals Program, via Reference (1). The PBNP reactor vessel internals program is based on Electric Power Research Institute (EPRI) Materials Reliability Program (MRP) technical report MRP-227, Pressurized Water Reactor Internals Inspection and Evaluation Guidelines, Revision 0.

Via References (2) and (3), the NRC determined additional information was required to enable the staff's continued review of the PBNP Reactor Vessel Internals Program. Enclosure 1 contains the NextEra response to the NRC staff's request for additional information.

This letter contains no new Regulatory Commitments and no revisions to existing Regulatory Commitments.

In accordance with 10 CFR 50.91, a copy of this letter is being provided to the designated Wisconsin Official.

I declare under penalty of perjury that the foregoing is true and correct.
Executed on August 16, 2012.

Very truly yours,

NextEra Energy Point Beach, LLC

Larry Meyer
Site Vice President

Enclosure

cc: Administrator, Region III, USNRC
Project Manager, Point Beach Nuclear Plant, USNRC
Resident Inspector, Point Beach Nuclear Plant, USNRC
PSCW

ENCLOSURE 1

NEXTERA ENERGY POINT BEACH, LLC
POINT BEACH NUCLEAR PLANT, UNITS 1 AND 2

REACTOR VESSEL INTERNALS INSPECTION PLAN
RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

The NRC staff determined that additional information was required (References 1 and 2) to enable the continued review of the Point Beach Nuclear Plant (PBNP) Reactor Vessel Internals Program (Reference 3). The following information is provided by NextEra Energy Point Beach, LLC (NextEra) in response to the NRC staff's request.

RAI-1

Applicant/Licensee Action Item 1 from the NRC staff's final safety evaluation (SE) of MRP-227-A, "Pressurized Water Reactor (PWR) Internals Inspection and Evaluation Guidelines," requires that applicants/licensees submit an evaluation that demonstrates that their plant is bounded by the assumptions regarding plant design and operating history that were made in the failure modes, effects and consequences analyses (FMECA) and functionality analyses for reactors of their design.

The licensee's response to Applicant/Licensee Action Item 1 in the RVI inspection plan addresses the core loading assumptions (switch to a low-leakage core) and operational (base loaded plant) aspects of design and operation that are mentioned in MRP-227-A, Section 2.4. An additional assumption listed in Section 2.4 of MRP-227-A is that there have been no design changes to the RVI beyond those identified in general industry guidance or recommended by the original vendors. Section 2.4 of MRP-227-A indicated that these assumptions are considered to represent any U.S PWR operating plant provided that these three assumptions are met, given the information on design and operation known to the MRP as of May 2007.

MRP-191, Revision 0, "Materials Reliability Program: Screening, Categorization and Ranking of Reactor Internals of Westinghouse and Combustion Engineering PWR [pressurized water reactor] Designs," (proprietary document), documents the screening for susceptibility to aging effects, the FMECA results, and the categorization and ranking of the RVI components. In addition to the assumptions listed in Section 2.4 of MRP-227-A, MRP-191 documents additional assumptions that were used. In particular, neutron fluence range, temperature, and material grade for each generic component of the Westinghouse design internals were used for input to the screening process. These values were determined based on an "expert elicitation" process. Stress values were not explicitly tabulated, but were recorded as either above the stress threshold (>30 ksi) or not based on the expert interviews.

MRP-232, Revision 0, "Materials Reliability Program: Aging Management Strategies for Westinghouse and Combustion Engineering PWR Internals," (proprietary document) reported more specific stress, temperature and neutron fluence values based on finite element analyses for selected high consequence of failure components identified in MRP-191.

The EPRI-MRP did not verify that the values of fluence, temperature, stress, and material, documented in MRP-191 and MRP-232 were bounding for all individual plants, and in fact MRP-227 states, "These evaluations were based on representative configurations and operational histories, which were generally conservative, but not necessarily bounding in every parameter."

The NRC staff expects that the licensee should have access to design information enabling verification that the material for each RVI component is bounded by the design assumptions of the MRP. In this context, the NRC staff requests that the licensee provide the following information:

(1) Describe the process used to verify that the RVI components at PBNP, Units 1 and 2 are bounded by the assumptions regarding the variable (i.e., neutron fluence, temperature, stress values, and materials) that were made for each component in the FMECA and functionality analyses supporting the development of MRP-227-A.

(2) To provide reasonable assurance that the RVI components are bounded by assumptions in the FMECA and functionality analyses supporting the development of MRP-227-A, the licensee is requested to respond to either part a) or part b) of this RAI:

(a) Provide the plant-specific values of neutron fluence (n/cm^2, E>1.0 MeV), temperature, stress, and materials for a sample of RVI components. The components selected should represent a range of neutron fluences, and temperatures. This information should identify whether the stress is greater or less than 30 ksi. Values of neutron fluence and temperature may be estimated or analytical values. The values should be the peak values of each parameter for each component (e.g., peak end-of-life value for fluence). Provide the method used to estimate the values, or describe the analysis method. An acceptable sample of components is:

i) Lower Core Plate

ii) Core Barrel Flange

iii) Barrel-Former Bolts

iv) Upper Core Barrel Welds

v) Lower Core Barrel Welds

vi) Upper Core Plate Alignment Pins

(b) Provide a qualitative assessment regarding the differences between the plant-specific variables (neutron fluence, temperature, stress values, and materials) and the variables of a "representative" PWR vessel used in developing the MRP-227-A report, for those components listed in part a) or for those components that are either identified as "Expansion" or were scoped out in the FMECA.

(3) If there are any components at PBNP, Units 1 and 2 not bounded by assumptions regarding neutron fluence, temperature, stress or material used in the development of MRP-227-A, describe how the differences were addressed in the plant-specific RVI Inspection Plan. The NRC staff requests that the licensee, as a part of its

demonstration, discuss whether there would be any changes to the screening, categorization, FMECA process and functionality analyses if the plant-specific variables (the neutron fluence, temperature, stress values, plant-specific operating experience, and materials) are used. This evaluation should address whether additional aging mechanisms would become applicable to the component.

(4) For any non-bounded components, determine if any changes to the inspection requirements of MRP-227-A are needed. Provide either plant-specific inspection requirements, an alternate aging management program (AMP), or if no changes to the inspection requirements are proposed, provide a justification for the adequacy of the existing MRP-227-A inspections for the unbounded components.

NextEra Response

Item (1)

NextEra has confirmed information regarding plant-specific reactor vessel internals (RVI) components for PBNP. The results of the comparison of the PBNP-specific components to those components used in the generic failure modes, effects and consequences analyses (FMECA) and functionality analysis (Materials Reliability Program (MRP)-191, Screening, Categorization, and Ranking of Reactor Internals Components for Westinghouse and Combustion Engineering PWR Design, Table 4-4) are provided below.

The process used to verify that the RVI components at PBNP are bounded by the assumptions regarding neutron fluence, temperature, stress values, and materials in the FMECA and functionality analyses is as follows:

1. Identification of typical Westinghouse Pressurized Water Reactor (PWR) RVI components.

2. Identification of PBNP PWR RVI components.

3. Comparison of the typical Westinghouse PWR RVI components to the PBNP PWR RVI components. No atypical items were identified by this comparison.

4. The materials are identified in Westinghouse letter WEP-02-58, Point Beach Units 1 and 2 Reactor Internals CMTR Summary, dated September 12, 2002, and are consistent with those materials identified in MRP-191 and MRP-232, Aging Management Strategies for Westinghouse and Combustion Engineering PWR Internals.

5. No modifications to the PBNP RVI have been made over the lifetime of the plant, except for those recommended by the original equipment manufacturer (OEM). Therefore, the assumptions regarding the neutron fluence, temperature, stress values, and materials used in performing the generic FMECA and functionality analysis are applicable for PBNP.

6. PBNP used a low leakage fuel management strategy starting with fuel cycle eight for Unit 1 and fuel cycle six for Unit 2. The core loading pattern was changed significantly prior to 30 years of operation. PBNP has primarily operated under base load conditions over the life of the plant. The FMECA and functionality analyses for MRP-227-A, Pressurized Water Reactor Internals Inspection and Evaluation Guidelines, were based

on the assumption of 30 years of operation with high leakage core loading patterns followed by 30 years of low leakage core loading patterns. Therefore, PBNP is bounded by the assumptions in the MRP documents regarding fluence.

7. Early in operating life, PBNP followed load as needed depending upon Wisconsin Electric Power Company system requirements. Load following operations diminished in the late 1970's as system requirements changed. Since this time, PBNP has typically operated at a fixed power level. The resulting load follow cycles are well within the reactor vessel internal's design basis transient analysis. The existence of the load following history will not affect the component categorizations nor recommended inspections since PBNP has been operated within its design and licensing basis. In addition, other conservatisms exist in that PBNP only operated with a high leakage core for seven cycles on Unit 1 and five cycles on Unit 2, which is significantly less than the 30 years assumed in MRP-227-A. Therefore, PBNP is bounded by the assumptions in the MRP documents regarding operational parameters.

8. The PBNP reactor vessel materials operate at temperatures between T_{cold} and T_{hot} that have nominally been not less than 523°F for T_{cold} and not higher than 611°F for T_{hot}.

9. The Westinghouse design of the RVI has been maintained by the owner over the lifetime of the plant and is bounded by the stresses assumed in MRP-191 and MRP-232.

10. PBNP has made modifications to the RVI. These modifications were all performed with the involvement of Westinghouse, the RVI designer. MRP-227 states that the recommendations are applicable to all U.S. PWR operating plants as of May 2007 for the three designs considered. PBNP has not made any modifications to RVI components since May 2007 other than replacement of split pins for Unit 1 with an upgraded material in 2008. The split pin replacement was performed by Westinghouse in accordance with the recommended design for split pins. The modification has no impact on the applicability of MRP-227 and is an example of the PBNP proactive approach to managing aging RVI.

11. MRP-191 and MRP-232 was used to organize, characterize, and rank the RVI parts into the various groups for development of aging management strategies, including Existing, Primary, Expansion, and No Additional Measures.

Item (2)

The following table provides qualitative assessment of the input parameters used in the FMECA and functionality analyses for typical Westinghouse PWR RVI compared to those used for the PBNP RVI for the sample of components.

RAI Item	Description	Parameters							
		Neutron Fluence		Temperature		Stress		Materials	
		Typical Plant	PBNP	Typical Plant	PBNP	Typical Plant	PBNP	Typical Plant	PBNP
i	Lower Core Plate	Reference MRP-191, Table 4-6, Screening Input Parameters for Westinghouse-Designed Plants Estimated Fluence Range (n/cm^2, E > 1 MeV) 1×10^{22} to 5×10^{22}	Same as MRP-191, Table 4-6, Screening Input Parameters for Westinghouse-Designed Plants Estimated Fluence Range (n/cm^2, E > 1 MeV) 1×10^{22} to 5×10^{22} bounds PBNP	Reference MRP-191, Table 4-6, Screening Input Parameters for Westinghouse-Designed Plants > 608°F	Same as MRP-191, Table 4-6, Screening Input Parameters for Westinghouse-Designed Plants > 608°F	Reference MRP-191, Table A-1, Results of Parameter Screening and Interviews with Analysts—Westinghouse Reactor Internals Effective Stress ≥ 30 ksi	Same as MRP-191, Table A-1, Results of Parameter Screening and Interviews with Analysts—Westinghouse Reactor Internals Effective Stress ≥ 30 ksi	Reference MRP-191, Table 4-6, Screening Input Parameters for Westinghouse-Designed Plants 304 SS	Reference Westinghouse Letter WEP-02-58 304 SS
ii	Core Barrel Flange	Reference MRP-191, Table 4-6, Screening Input Parameters for Westinghouse-Designed Plants Estimated Fluence Range (n/cm^2, E > 1 MeV) $< 10^{20}$	Same as MRP-191, Table 4-6, Screening Input Parameters for Westinghouse-Designed Plants Estimated Fluence Range (n/cm^2, E > 1 MeV) $< 10^{20}$ bounds PBNP	Reference MRP-191, Table 4-6, Screening Input Parameters for Westinghouse-Designed Plants T_{hot}	The PBNP reactor vessel materials operate at temperatures < 611°F (T_{hot}).	Reference MRP-191, Table A-1, Results of Parameter Screening and Interviews with Analysts—Westinghouse Reactor Internals Effective Stress < 30 ksi	Same as MRP-191, Table A-1, Results of Parameter Screening and Interviews with Analysts—Westinghouse Reactor Internals Effective Stress < 30 ksi	Reference MRP-191, Table 4-6, Screening Input Parameters for Westinghouse-Designed Plants 304 SS	Reference Westinghouse Letter WEP-02-58 304 SS

RAI Item	Description	Parameters							
		Neutron Fluence		Temperature		Stress		Materials	
		Typical Plant	PBNP	Typical Plant	PBNP	Typical Plant	PBNP	Typical Plant	PBNP
iii	Barrel-Former Bolts	Reference MRP-191, Table 4-6, Screening Input Parameters for Westinghouse-Designed Plants Estimated Fluence Range (n/cm^2, E > 1 MeV) 5 x 10^{22}	Same as MRP-191, Table 4-6, Screening Input Parameters for Westinghouse-Designed Plants Estimated Fluence Range (n/cm^2, E > 1 MeV) 5 x 10^{22} bounds PBNP	Reference MRP-191, Table 4-6, Screening Input Parameters for Westinghouse-Designed Plants > 608°F	Same as MRP-191, Table 4-6, Screening Input Parameters for Westinghouse-Designed Plants > 608°F	Reference MRP-191, Table A-1, Results of Parameter Screening and Interviews with Analysts—Westinghouse Reactor Internals Effective Stress ≥ 30 ksi	Same as MRP-191, Table A-1, Results of Parameter Screening and Interviews with Analysts—Westinghouse Reactor Internals Effective Stress ≥ 30 ksi	Reference MRP-191, Table 4-6, Screening Input Parameters for Westinghouse-Designed Plants 316 SS or 347 SS	Reference WCAP-13266, Revision 1 347 SS
iv	Upper Core Barrel Welds	Reference MRP-191, Table 4-6, Screening Input Parameters for Westinghouse-Designed Plants Estimated Fluence Range (n/cm^2, E > 1 MeV) 1 x 10^{21} to 1 x 10^{22}	Same as MRP-191, Table 4-6, Screening Input Parameters for Westinghouse-Designed Plants Estimated Fluence Range (n/cm^2, E > 1 MeV) 1 x 10^{21} to 1 x 10^{22} bounds PBNP	Reference MRP-191, Table 4-6, Screening Input Parameters for Westinghouse-Designed Plants T_{hot}	The PBNP reactor vessel materials operate at temperatures < 611°F (T_{hot}).	Reference MRP-191, Table A-1, Results of Parameter Screening and Interviews with Analysts—Westinghouse Reactor Internals Effective Stress ≥ 30 ksi	Same as MRP-191, Table A-1, Results of Parameter Screening and Interviews with Analysts—Westinghouse Reactor Internals Effective Stress ≥ 30 ksi	Reference MRP-191, Table 4-6, Screening Input Parameters for Westinghouse-Designed Plants 304 SS	Reference Westinghouse Letter WEP-02-58 304 SS

RAI Item	Description	Parameters							
		Neutron Fluence		Temperature		Stress		Materials	
		Typical Plant	PBNP	Typical Plant	PBNP	Typical Plant	PBNP	Typical Plant	PBNP
v	Lower Core Barrel Welds	Reference MRP-191, Table 4-6, Screening Input Parameters for Westinghouse-Designed Plants Estimated Fluence Range (n/cm^2, E > 1 MeV) 1×10^{21} to 1×10^{22}	Same as MRP-191, Table 4-6, Screening Input Parameters for Westinghouse-Designed Plants Estimated Fluence Range (n/cm^2, E > 1 MeV) 1×10^{21} to 1×10^{22} bounds PBNP	Reference MRP-191, Table 4-6, Screening Input Parameters for Westinghouse-Designed Plants. T_{cold}	The PBNP reactor vessel materials operate at temperatures ≥523°F (T_{cold}).	Reference MRP-191, Table A-1, Results of Parameter Screening and Interviews with Analysts—Westinghouse Reactor Internals Effective Stress ≥ 30 ksi	Same as MRP-191, Table A-1, Results of Parameter Screening and Interviews with Analysts—Westinghouse Reactor Internals Effective Stress ≥ 30 ksi	Reference MRP-191, Table 4-6, Screening Input Parameters for Westinghouse-Designed Plants 304 SS	Reference Westinghouse Letter WEP-02-58 304 SS
vi	Upper Core Plate Alignment Pins	Reference MRP-191, Table 4-6, Screening Input Parameters for Westinghouse-Designed Plants Estimated Fluence Range (n/cm^2, E > 1 MeV) 7×10^{20} to 1×10^{21}	Same as MRP-191, Table 4-6, Screening Input Parameters for Westinghouse-Designed Plants Estimated Fluence Range (n/cm^2, E > 1 MeV) 7×10^{20} to 1×10^{21} bounds PBNP	Reference MRP-191, Table 4-6, Screening Input Parameters for Westinghouse-Designed Plants. T_{hot}	The PBNP reactor vessel materials operate at temperatures < 611°F (T_{hot}).	Reference MRP-191, Table A-1, Results of Parameter Screening and Interviews with Analysts—Westinghouse Reactor Internals Effective Stress < 30 ksi	Same as MRP-191, Table A-1, Results of Parameter Screening and Interviews with Analysts—Westinghouse Reactor Internals Effective Stress < 30 ksi	Reference MRP-191 Table 4-6, Screening Input Parameters for Westinghouse-Designed Plants 304 SS	Reference Westinghouse Letter WEP-02-58 304 SS

General Notes

1) The PBNP reactor vessel materials operate at temperatures between T_{cold} and T_{hot} that have nominally been not less than 523°F for T_{cold} and not higher than 611°F for T_{hot}. The design temperature for the PBNP reactor vessel is 650°F.
2) Criteria for material, temperature, and fluence are listed in MRP-191, Table 4-6, Screening Input Parameters for Westinghouse-Designed Plants.
3) Criteria for stress is depicted in MRP-191, Table 3-2, Irradiation Assisted Stress Corrosion Cracking (IASCC) Screening Criteria and MRP-191, Figure 3-1, MRP-175 Screening Criteria for IASCC.
4) Criteria for SCC is listed in MRP-191, Table 3-1, Stress Corrosion Cracking (SCC) Screening Criteria for PWR Internals Materials.

Item (3)

The PBNP RVI components are bounded by assumptions regarding neutron fluence, temperature, stress and material used in the development of MRP-227-A. Westinghouse letter WEP-02-58 lists the materials for PBNP RVI components. The PBNP components, materials, and design are consistent with those identified as Typical Westinghouse PWR Internals Components. The PBNP reactor vessel materials operate at temperatures between T_{cold} and T_{hot} that have nominally been not less than 523°F for T_{cold} and not higher than 611°F for T_{hot}. Because the design of the RVIs has been maintained over the lifetime of the plant and the materials are the same as those identified on MRP-191, MRP-232, and MRP-227-A, the stress values are bounded by the assumptions contained in MRP-191, MRP-232, and MRP-227-A.

Item (4)

The PBNP RVI components are bounded by the Typical Westinghouse PWR internals components outlined in MRP-227-A and the applicable referenced documents, including MRP-191 and MRP-232. The PBNP reactor vessel internals inspection program was written to comply with MRP-227-A. No changes in the inspection requirements are being proposed at this time in that the PBNP inspection program complies with MRP-227-A as indicated above.

RAI-2

The licensee has listed several plant-specific examples of operating experience related to the aging degradation in the RVI components. The staff requests more specific information related to these examples. For the baffle-former bolts that were examined in 1997, what was the total number of baffle-former bolts in Unit 2, how many bolts were examined, and how many were cracked? Did the cracked bolts conform to any pattern related to neutron exposure? How many were replaced, and did the replacement guarantee the structural margins? What was the original material and the replacement material in 1997? Describe the planned inspections. Is the inspection procedure that is planned to be used in the future at PBNP the same as that used in the 1997 bolt inspections?

MRP-51 lists in Table 2-10 and 2-14 the measured irradiation conditions, temperature and fluence, for samples taken from the 1997 inspections. Given these values for the irradiation conditions in 1997, provide estimated irradiation conditions for the most susceptible baffle-former bolts from Unit 1 in 2013, and estimated irradiation conditions for 2015 that represent the most susceptible of the Unit 2 bolts that were replaced in 1997 as well as the most susceptible of the Unit 2 bolts that were not replaced in 1997.

NextEra Response

Specific information for the Unit 2 baffle-former bolts examination is contained in the 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.

This report discusses the baffle bolt inspection and replacement processes for the two-loop plants, PBNP Unit 2 and R. E. Ginna Nuclear Power Plant (Ginna). The deployment of the processes and equipment covered in this report occurred at PBNP Unit 2 during late December 1998 and early January 1999. The ultrasonic inspection activity at PBNP Unit 2 involved all 728 baffle to former bolts. Investigators removed a total of 176 bolts and replaced them with 175 improved design bolts. They destructively tensile tested a total of 146 of the 176 removed bolts and sent 12 bolts to the Westinghouse hot cell for evaluation.

The ultrasonic evaluation of baffle to former bolts identified a total of 55 bolts with 114 ultrasonic testing (UT) indications at PBNP Unit 2 and 59 at Ginna. Of the 114 PBNP indications, approximately 14 were substantiated failures by virtue of their heads separating from their shanks during the removal process or at low loads associated with subsequent mechanical testing. Bolts from both plants, some of which contained UT indications, were sent to the hot cell for further analyses. The most likely cause of failures appeared to be a combination of Irradiation Assisted Stress Corrosion Cracking (IASCC) and cyclic stress. The indications did not conform to any pattern related to neutron exposure.

There are 728 baffle plate to former bolts in the two-loop Westinghouse designed vessel with 104 bolts at each former plate elevation. The original bolt design conforms to the standard ANSI B18.3 for socket head cap screws except that the shank is machined to a diameter smaller than that specified by the standard. The bolt material is AISI Type 347 stainless steel with threads that are rolled and chrome plated. Each bolt is installed in a counter-bore in the baffle plate and tightened. A washer is inserted in the counter-bore over the bolt head and tack welded to the baffle plate at the washer OD and also tack welded to the bolt head at the washer ID. These bolts assure that the RVI maintain structural integrity in the core region during normal operating and upset conditions such that the plant can safely shutdown.

The replacement bolts were fabricated from SA-193 Class 2 B8M (strain-hardened Type 316) material in accordance with ASME Section III, NG-2000 and Code Case N-60-4. Since the Owner's analysis assumed that each bolt had a minimum yield strength of 80 ksi and Code Case N-60-4 limits the yield strength of strain-hardened fastener material to 90 ksi, each bar from which the bolts were made was tested for mechanical properties. Only those bars with yield strengths between 80 and 90 ksi, inclusive, were used to fabricate the replacement baffle-to-former plate bolts.

Significant additional controls were imposed on the material and additional testing was performed in order to provide optimum material properties and resistance to stress corrosion cracking. These include material chemistry control, refined grain size requirements, and an ASTM A262 Practice E susceptibility to intergranular corrosion test (with material in the sensitized condition).

The replacement bolt is functionally the same as the original bolt in that the threads and thread length, shank diameter, and head diameter are essentially unchanged, except that the threads on the replacement bolt were not chrome plated as were the original bolts. Presumably, the purpose of the chrome plating on the original bolts was to prevent galling during installation and serves no in-service function. Some differences exist in the head of the replacement bolt. These differences exist to accommodate the locking device and installation tooling, and to enhance the ability of the bolt to be ultrasonically examined after being placed in service. None of these changes affect the functionality of the bolt while in service.

Several design enhancements were employed on the replacement bolt to reduce the susceptibility of the bolt to Primary Water Stress Corrosion Cracking (PWSCC) and IASCC. The head to shank transition radius was changed from a simple radius on the original bolt to a compound (elliptical) radius on the replacement bolt. This has the effect of reducing stress concentrations in this region of the bolt, thereby reducing the susceptibility of the bolt to cracking in this region. Additionally, the surface of the bolt from the OD of the bolt head seating surface to the shank-to-thread transition was shot peened to induce residual compressive stresses in the surface of the bolt, which also reduces the susceptibility of the bolt to stress corrosion cracking. Residual compressive stresses were induced in the bolt's threaded region during the cold rolling process.

Testing was performed so that the preload in the original bolts could be matched with the replacement bolts. Ten original bolts (with chrome plated threads) and ten replacement bolts were fabricated and were tested for torque versus preload characteristics in their respective installation conditions (i.e., the original bolts were tested in an air environment and the replacement bolts were tested in a water environment). The preload generated in the original bolt by the specified installation torque was determined, and then the torque required to achieve the same preload in the replacement bolts was determined.

The planned UT examinations for MRP-227-A "Primary" components include all 728 baffle plate to former bolts for each unit. The examination method (UT) is the same as used previously for Unit 2. Improvements to the UT equipment have been made to address issues identified during the initial examinations of the baffle plate to former bolts.

Unit 1 will be examined in 2013 with an operating time of about 34 EFPY and exposure of 73.2 displacements per atom (dpa). Similarly, Unit 2 will be examined in 2015 with an operating time of around 36 EFPY and exposure of 77.7 dpa. Table 2-14 of MRP-51, Hot Cell Testing of Baffle/Former Bolts Removed from Two Lead Plants, lists the Total Neutron Exposure for PBNP Slow Strain Rate Test (SSRT) specimens as between 3 and 15 dpa. Tables 5.1.2-7 and 5.1.2-8 of WCAP-16983-P, Point Beach Units 1 and 2 Extended Power Uprate (EPU) Engineering Report, list the maximum neutron exposure received by reactor internals baffle plates for PBNP Units 1 and 2, respectively, as a function of reactor operating time.

RAI-3

The NRC staff requests the license discuss the results from the existing programs (Attachment D) and the extent of aging degradation (if any) that occurred thus far in the following components at PBNP:

(a) baffle-edge bolts,

(b) clevis insert bolts,

(c) flux thimble tubes,

(d) core barrel bolting, and

(e) thermal shields.

NextEra Response

Components in the existing programs (Attachment D) were examined per the 10-Year Inservice Inspections during U1R32 (Spring 2010) and U2R30 (Fall 2009). No recordable indications were noted during the VT-3 examinations of these components.

The extent of aging degradation which has occurred thus far in the components listed below at PBNP is as follows:

(a) Baffle-Edge Bolts: VT-3 examinations of the interior of the lower core barrel, which included the area with the baffle-edge bolts, were performed during the 10-Year Inservice Inspections of the Reactor Vessel. No broken bolts or tack welds were noted.

(b) Clevis Insert Bolts: VT-3 examinations of the interior of the reactor vessel, which included the area with the clevis insert keys, were performed during the 10-Year Inservice Inspections of the Reactor Vessel. No broken bolts or tack welds were noted.

(c) Flux Thimble Tubes: The PBNP Thimble Tube Condition Assessment Program describes the eddy current examination history of the flux thimble tubes. The original incore thimble tubes were replaced in both Unit 1 and Unit 2 in 1985. It was necessary to replace these tubes due to internal blockages. No leaking incore thimble tubes were ever discovered during the first 13 years of operation. The replacement tubes are made of stainless steel Type 316 with a nominal outer diameter of 0.313" and a nominal inside diameter of 0.210". The original tubes measure 0.300" O.D. and 0.200" I.D. The additional size was used to prevent blockages. Five thimble tubes in Unit 1 were replaced in 1998.

(d) Core Barrel Bolting: Core Barrel Bolting does not apply to Westinghouse internals.

(e) Thermal Shields: VT-3 examinations of the exterior of the lower core barrel, which includes the thermal shield, were performed during the 10-Year Inservice Inspections of the Reactor Vessel. No indications were noted.

RAI-4

Historically, the following materials used in the PWR RVI components were known to be susceptible to some of the aging degradation mechanisms that are identified in the MRP-227-A report. In this context, the NRC staff requests that the licensee confirm that these materials are not currently used in the RVI components at PBNP. If they are used in any PBNP RVI components, please identify any service history associated with the components that is not captured in the response to RAI 2.

(1) Nickel base alloys-Inconel 600; Weld Metals-Alloy 82 and 182 and Alloy X-750

(2) Alloy A-286 ASTM A 453 Grade 660, Condition A or B

(3) Stainless steel type 347 material (excluding baffle-former bolts)

(4) Precipitation hardened (PH) stainless steel materials—17-4 and 15-5

(5) Type 431 stainless steel material

NextEra Response

Materials used in fabrication of the PBNP RVI are listed in Westinghouse letter WEP-02-58. PBNP reviewed the contents of this document for the listed materials. The review results are as follows.

(1) The following PBNP reactor internals are fabricated from Nickel base alloys (Inconel 600, Weld Metals - Alloy 82 and 182, and Alloy X-750):

- Clevis Insert Locking Mechanisms (ASTM B-166 (Inconel 600))
- Clevis Insert Bolts (Inconel X-750)

The clevis insert locking mechanisms and bolts were VT-3 examined during the 10-Year Inservice Inspections during U1R32 (Spring 2010) and U2R30 (Fall 2009). No broken locking mechanisms, bolts, or tack welds were identified during the VT-3 examinations.

(2) No RVI components are listed as being fabricated from Alloy A-286 or ASTM A-453 Grade 660, Condition A or B.

(3) No RVI components (excluding baffle-former bolts and barrel-former bolts) are listed as being fabricated from stainless steel Type 347 material.

(4) No RVI components are listed as being fabricated from precipitation hardened (PH) stainless steel materials - 17-4 and 15-5.

(5) No RVI components are listed as being fabricated from Type 431 stainless steel materials.

RAI-5

MRP-227-A provides general descriptions for the examination coverage for several primary components where the licensee must make specific decisions to complete the required exams. For example, the licensee is required to inspect 20% of the CRGT guide card assemblies per Attachment B of the submittal. For the CRGT lower flange welds, 100% of outer (accessible) CRGT lower flange weld surfaces and adjacent base metal on the individual periphery CRGT assemblies is the exam coverage. For the core barrel assembly welds, MRP-227-A requires 100% of one side of the accessible surfaces. The NRC staff requests that the licensee provide specifics on what will be included in the examination with an explanation for the selection process, which should include the following aspects:

(1) most susceptible areas to experience aging degradation,

(2) high stress areas,

(3) accessibility issues and,

(4) plant-specific operating experience.

NextEra Response

NextEra plans VT-3 examinations of 100% of the Control Rod Guide Tube (CRGT) guide card assemblies for both Units, based on the recommendations contained in WCAP-17020-P, Point Beach Unit 1 Upper Internal Guide Tube – Guide Card Wear Evaluation. Some guide card wear was noted during the inspections performed for Unit 1 in 2008.

For CRGT lower flange welds, 100% of outer (accessible) CRGT lower flange weld surfaces and adjacent base metal on the individual periphery CRGT assemblies will be examined (EVT-1) per MRP-227-A guidance:

(1) Most Susceptible Areas to Experience Aging Degradation: Lower flange welds which connect the flange to the vertical sheaths and C-tubes of the assembly, which are "Primary" components per MRP-227-A.

(2) High-Stress Areas: The welds on the periphery CRGT assemblies, which are being examined based on MRP-227-A guidance, should have similar stresses as the welds on the interior CRGT assemblies.

(3) Accessibility Issues: Portions of the welds are difficult to access for EVT-1 inspection due to minimal gaps around the CRGT assemblies.

(4) Plant-specific Operating Experience: No degradation has been noted during previous VT-3 examinations of the CRGTs.

For the core barrel assembly welds, 100% of one side of the accessible surfaces will be examined (EVT-1) per MRP-227-A guidance:

(1) Most Susceptible Areas to Experience Aging Degradation: Core barrel upper and lower flange welds, along with core barrel upper and lower girth welds, which are "Primary" components per MRP-227-A.

(2) High-Stress Areas: The welds on either side of the core barrel, which are being examined based on MRP-227-A guidance, should have similar stresses as the welds on the other side of the core barrel.

(3) Accessibility Issues: Some welds are only accessible with a core barrel pull. The lower core barrel cylinder girth weld has limited accessibility due to the baffle assembly on the ID of the core barrel and a minimal gap between the core barrel and thermal shield on the OD.

(4) Plant-specific Operating Experience: No degradation has been noted during previous VT-3 examinations of the core barrel.

RAI-6

In Attachment D for existing programs in the December 15, 2011, submittal, the licensee stated that the flux thimble tubes were last examined in the spring of 2010 for Unit 1 and the fall of 2009 for Unit 2, referencing NUREG-1801, Revision 1. The staff does not see any specific guidance in the reference for selecting the inspection frequency. What is the inspection frequency for the flux thimble tubes at Point Beach, Units 1 and 2?

The staff also requests that the licensee update their aging management program (AMP) to reference NUREG-1801, Revision 2 (GALL) and describe how their plant-specific AMP, LR-AMP-006-TTI, "Thimble Tube Inspection Program Basis Document" compares to the XI.M37, "Flux Thimble Tube Inspection" AMP in the GALL report.

NextEra Response

The PBNP Thimble Tube Condition Assessment Program describes the eddy current inspection frequency of the flux thimble tubes. The inspection frequency is based on the maximum wall loss noted in a region of active fretting and the projected wear which would occur based on a known wear rate. To ensure conservative testing intervals, inspections are normally conducted in the outage prior to the outage that would be identified as the one before the lowest minimum thimble tube life calculation, or at least every six years.

Although PBNP is licensed to the initial Generic Aging Lessons Learned (GALL) report, NextEra has initiated a change request to update the Thimble Tube Inspection Program Basis Document for License Renewal, LR-AMP-006-TTI, adding reference to NUREG-1801, Generic Aging Lessons Learned (GALL) Report, Revision 2, and a description of how LR-AMP-006-TTI compares to the GALL Aging Management Program XI.M37, Flux Thimble Tube Inspection.

RAI-7

MRP-227-A, Table 3-3, indicates that control rod guide tube support pins (commonly referred to as split pins), Alloy X-750, are classified under the "Existing Inspection Program." That would suggest that they are ASME Section XI, Examination Category B-N-3 components. However, the split pins are NOT included in the existing inspection program Table 4-9 in MRP-227-A or Attachment D in the PBNP inspection plan.

MRP-227-A, Section 4.4.3, states that the performance of the split pins should follow the supplier recommendations. However, the guideline included in Section 3.2.5.3 of the staff's safety evaluation for the MRP-227-A report is more specific and summarized below.

Westinghouse guide tube support pins are made from either 316 stainless steel or Alloy X750. There have been issues with cracking of the original Alloy X750 pins that were originally discovered during the ASME Section XI B-N-3 inspections. Many licensees have replaced them with type 316 stainless steel materials. Licensees shall evaluate the adequacy of their plant-specific existing program and ensure that the aging degradation is adequately managed during the extended period of operation regardless of the material used for the guide tube support pins (split pins). Therefore, the NRC staff recommends that the evaluation consider the need to replace the Alloy X750 support pins (split pins), if applicable, or inspect the replacement type 316 stainless steel support pins (split pins) to ensure that cracking has been mitigated and that aging degradation due to any of the potential mechanisms is adequately monitored during the extended period of operation.

In the December 19, 2011, submittal, NextEra stated that the split pins were replaced with more SCC resistant 316 stainless steel material. The licensee did not indicate that it will inspect these pins during the license renewal period. Therefore, the NRC staff requests that the NextEra provide an explanation for not performing routine ASME Code, Section XI, inspections on the 316 stainless steel split pins.

NextEra Response

While the split pins are not categorized as an ASME Section XI Code Item, a VT-3 inspection has historically been performed on accessible portions of accessible split pins when other items of the reactor internals have been inspected for ASME Section XI. Since failure of a split pin could have significant economic impact to a nuclear site, NextEra will continue to perform a VT-3 inspection on accessible portions of accessible split pins concurrent with the B-N-3 inspection when the core barrel is removed from the reactor vessel.

References

(1) NRC electronic mail to NextEra Energy Point Beach, LLC, dated June 7, 2012, Point Beach Units 1 and 2 – Draft RAI on the Reactor Vessel Internals Inspection Plan (TAC ME8235 and ME8236) (ML12159A113)

(2) NRC electronic mail to NextEra Energy Point Beach, LLC, dated July 10, 2012, Point Beach Units 1 and 2 – Draft Request for Additional Information re: Reactor Vessel Internals Inspection Plan (TAC Nos. ME8235 and ME8235) (ML12198A050)

(3) NextEra Energy Point Beach, LLC letter to NRC, dated December 19, 2011, License Renewal Commitment, Reactor Vessel Internals Program Submittal (ML113540301)