



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

December 10, 2014

Mr. Eric McCartney  
Site Vice President  
NextEra Energy Point Beach, LLC  
Point Beach Nuclear Plant  
6610 Nuclear Road  
Two Rivers, WI 54241-9516

**SUBJECT: POINT BEACH NUCLEAR PLANT, UNIT 1 – RELIEF REQUEST 1-RR-6,  
PROPOSED ALTERNATIVE FROM THE REQUIREMENTS OF THE  
AMERICAN SOCIETY OF MECHANICAL ENGINEERS BOILER AND  
PRESSURE VESSEL CODE FOR RE-EXAMINATION OF REACTOR  
PRESSURE VESSEL “A” INLET NOZZLE WELD (TAC NO. MF3318)**

Dear Mr. McCartney:

By letter dated December 30, 2013, NextEra Energy Point Beach, LLC (NextEra) submitted a request to the U.S. Nuclear Regulatory Commission (NRC) for relief from certain requirements specified in the American Society of Mechanical Engineers *Boiler and Pressure Vessel Code* (ASME Code), Section XI, at the Point Beach Nuclear Plant (Point Beach), Unit 1.

Specifically, pursuant to Title 10 of the *Code of Federal Regulations* (10 CFR), 50.55a(a)(3)(i), NextEra submitted Relief Request 1-RR-6, requesting deferral of the inspection of the Point Beach, Unit 1, reactor pressure vessel (RPV) “A” inlet nozzle weld RC-32-MRCL-AIII-03 from 2017 to 2020. NextEra requested that the second re-examination of the weld be waived until the next scheduled 10-year inservice inspection interval of the RPV due, in part, to radiological and industrial safety concerns involved in core barrel removal, and on the basis that the proposed alternative continues to provide an acceptable level of quality and safety.

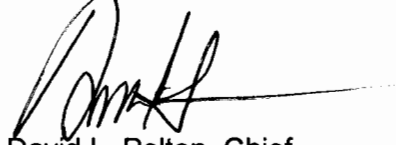
The NRC staff has reviewed the subject request and concludes, as set forth in the enclosed safety evaluation, that NextEra has adequately addressed all of the regulatory requirements set forth in 10 CFR 50.55a(a)(3)(i). Therefore, the NRC determined that complying with the specified requirement would provide an acceptable level of quality and safety. Furthermore, the staff determines that authorizing the use of the alternative provides reasonable assurance of structural integrity of the subject components.

E. McCartney

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If you have any questions, please contact Terry Beltz at (301) 415-3049, or via e-mail at Terry.Beltz@nrc.gov.

Sincerely,

A handwritten signature in black ink, appearing to read 'D. Pelton', with a long horizontal line extending to the right.

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

Docket Nos. 50-266

Enclosure:  
Safety Evaluation

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UNITED STATES  
NUCLEAR REGULATORY COMMISSION  
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SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

REGARDING RELIEF REQUEST 1-RR-6

FOR THE FIFTH 10-YEAR INSERVICE INSPECTION INTERVAL

NEXTERA ENERGY POINT BEACH, LLC

POINT BEACH NUCLEAR PLANT, UNIT 1

DOCKET NO. 50-266

(TAC NO. MF3318)

1.0 INTRODUCTION

By letter dated December 30, 2013 (Agencywide Documents Access and Management System (ADAMS) Package Accession No. ML14006A317), NextEra Energy Point Beach, LLC (NextEra, the licensee) requested relief from certain requirements of the American Society of Mechanical Engineers *Boiler and Pressure Vessel Code* (ASME Code) at the Point Beach Nuclear Plant (Point Beach), Unit 1. The proposed alternative is documented in Relief Request 1-RR-6 for deferral of inspection of the reactor pressure vessel (RPV) "A" inlet nozzle weld from 2017 to 2020.

Specifically, pursuant to Title 10 of the *Code of Federal Regulations* (10 CFR) 50.55a(a)(3)(i), the licensee requested re-examination of the Point Beach, Unit 1, RPV "A" inlet nozzle weld be waived until the next scheduled 10-year inservice inspection (ISI) interval of the RPV based, in part, on radiological and industrial safety concerns involved in core barrel removal, and that the proposed alternative provides an acceptable level of quality and safety.

2.0 REGULATORY EVALUATION

Pursuant to 10 CFR 50.55a(g)(4), ASME Code Class 1, 2, and 3 components (including supports) must meet the requirements, except the design and access provisions and the pre-service examination requirements, set forth in the ASME Code, Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components," to the extent practical within the limitations of component design, geometry, and materials of construction. The regulations require that in-service examination of components and system pressure tests conducted during the first 10-year interval and subsequent intervals comply with the requirements in the latest edition and addenda of Section XI of the ASME Code, incorporated by reference in 10 CFR 50.55a(b), 12 months prior to the start of the 120-month interval, and subject to the limitations and modifications listed therein. The Code of record for the current fifth 10-year ISI

Enclosure

interval at Point Beach, Unit 1, is the 2007 Edition up to and including the 2008 Addenda of the ASME Code, Section XI.

The regulation in 10 CFR 50.55a(a)(3) states, in part, that alternatives to the requirements of paragraph (g) of 10 CFR 50.55a may be used, when authorized by the NRC, if the licensee demonstrates (i) the proposed alternatives would provide an acceptable level of quality and safety or (ii) compliance with the specified requirements would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety.

Based on the above, and subject to the following technical evaluation, the NRC staff finds that regulatory authority exists for the licensee to request relief, and that the regulatory authority exists to authorize the proposed alternative pursuant to 10 CFR 50.53a(a)(3).

### 3.0 TECHNICAL EVALUATION

#### 3.1 ASME Code Components Affected

- Component: Reactor Coolant System Loop "A" Cold Leg Stainless Steel Elbow to Inlet Nozzle Weld at 328.5°
- Weld Number: RC-32-MRCL-AIII-03
- ASME Code Class: Code Class 1
- Examination Category: R-A (B-F)
- Item Number: R1.20 (B5.10)

#### 3.2 Applicable Code Edition and Addenda

The Code of Record for the current fifth 10-year ISI interval at Point Beach, Unit 1, which began on July 30, 2012, and ends on July 30, 2022, is ASME Code, Section XI, 2007 Edition with Addenda through 2008.

#### 3.3 Applicable Code Requirement

ASME Code, IWB-2420(b) states, in part, that "if a component is accepted for continued service in accordance with IWB-3132.3 or IWB-3142.4, the areas containing flaws or relevant conditions shall be re-examined during the next three inspection periods listed in the schedule of the inspection program of IWB-2400."

#### 3.4 Proposed Alternative

The licensee proposes not to perform the second successive ultrasonic and eddy current re-examination per IWB-2420(b) during the 2nd period of the 5<sup>th</sup> interval. VT-2 visual examinations are performed during the Class 1 system leakage test at the end of each refueling outage. The licensee is requesting that the second re-examination of the flaw in question be waived until the next scheduled 10-year ISI of the RPV (currently scheduled in 2020).

### 3.5 Licensee's Basis for Use of the Proposed Alternative

During Unit 1 refueling outage 32 in the spring of 2010, the licensee performed a phased-array ultrasonic (PA-UT) examination of the reactor vessel inlet nozzle-to-pipe weld (RC-32-MRCL-AIII-03) and identified an ASME Section XI Code rejectable indication. The weld is a dissimilar metal weld (cast stainless elbow with stainless weld and stainless buttering). The indication was recorded 18 inches from top dead center and 2.1 inches from the weld centerline on the nozzle side of the weld in the nozzle forging, and approximately 0.9 inches from the buttering.

The licensee notes that the indication could be seen in the "toward", "away", "clockwise", and "counterclockwise" directions, indicating that it is volumetric in nature (e.g., slag inclusion). In addition, the licensee performed an eddy current examination to confirm that the indication was not connected to the inside wetted surface. The indication was found to be acceptable for further service without repair for the remainder of the life of Unit 1, including the period of extended operation, using the acceptance criteria found in ASME Section XI, paragraph IWB-3600, and was re-scheduled for examination in the three subsequent, successive inspection periods in accordance with IWB-2420(b).

During Unit 1 refueling outage 34 in the spring of 2013, the licensee performed the first successive examination using identical techniques (PA-UT), supplemented by eddy current testing. This examination confirmed that the indication had not changed in size and that the indication was not connected to the inside surface.

The licensee further stated that performance of the PA-UT and eddy current inspections requires access to the inside diameter of the pipe. These inspections are normally performed when the core barrel is removed to facilitate access to the RPV inlet piping. Removal of the core barrel is a significant operation that has radiological and industrial safety concerns. Since the flaw has remained essentially unchanged from the first examination, the licensee believes that performing the VT-2 visual examinations during the Class 1 system leakage test, with acceptable results, provides reasonable assurance of continued structural integrity of the subject component. Therefore, the licensee contends that an acceptable level of quality and safety is maintained until a PA-UT inspection will be performed during the third period of the fifth 10-year ISI interval coincident with the scheduled RPV inspection in 2020.

### 3.6 Duration of Proposed Alternative (as stated by the licensee)

The proposed alternative will be used for the fifth 10-year ISI interval of the inservice inspection program for Point Beach, Unit 1, which is scheduled to end on July 30, 2022.

### 3.7 NRC Staff Evaluation

During the Unit 1 refueling outage in the spring of 2010, the licensee detected a circumferential indication in the stainless steel safe end-to-elbow weld RC-32-MRCL-AIII-03 of the RPV "A" cold leg nozzle. The flaw length was 0.71 inches with a depth of 0.692 inches. The weld thickness is 3.27 inches. The flaw is approximately 19 percent through-wall, which exceeds the acceptance standards of the ASME Code, Section XI, IWB-3514. In accordance with IWB-3142.4, the licensee performed a flaw evaluation to accept the flaw for continued service in accordance with the ASME Code, Section XI, IWB-3640, as documented in Westinghouse Report LTR-PAFM-10-50-NP, Revision 0, "Section XI Flaw Evaluation of Indication Recorded

on RC-32-MRCL-AIII-03 of the Point Beach Unit 1 Inlet Nozzle Pipe Weld,” provided as Attachment 2 of the licensee’s submittal (ADAMS Accession No. ML14006A319).

The ASME Code, Section XI, IWB-3142.4 requires that “...[a] component accepted for continued service based on analytical evaluation shall be subsequently examined in accordance with IWB-2420(b) and (c)...” In accordance with the ASME Code, Section XI, IWB-2420(b), the licensee needs to examine the indication during each of the three subsequent, successive inspection periods.

The licensee completed the first successive examination in 2013 during refueling outage 34, and confirmed that the flaw did not grow. The second successive examination should be performed prior to the end of the second period of the Unit 1 fifth 10-year ISI interval, in the fall of 2017. In lieu of performing the second successive examination, the licensee requested to defer the examination to correspond with the next scheduled ISI of the RPV which will facilitate removal of the vessel core barrel for the current fifth 10-year ISI interval. The licensee based its request, in part, on the results of the flaw evaluation and ultrasonic and eddy current examinations performed in 2010 and 2013.

The NRC staff reviewed the Westinghouse Report LTR-PAFM-10-50-NP, Revision 0, which assumed the fatigue degradation mechanism to predict slow crack growth for the indication in weld RC-32-MRCL-AIII-03. Given that the flaw is not connected to the wetted surface, the staff agrees that the fatigue is the most likely degradation growth mechanism. The staff also notes that, as presented in LTR-PAFM-10-50-NP, Revision 0, the crack growth due to thermal fatigue is small for a 40-year period. These results are consistent with numerous other embedded flaw analyses performed for indications in similar primary coolant piping locations. Therefore, the NRC staff finds that the crack growth for an approximate 7-year period (2013 to 2020) during which the weld is not examined by ultrasonic examination will be minimal.

As reported in the relief request, the indication is approximately 19 percent through-wall (crack depth). The NRC staff notes that the maximum allowable depth for the circumferential flaw is 75 percent through-wall, in accordance with the ASME Code, Section XI. In the highly unlikely case that the 0.71 inch flaw could grow to be a continuous 360 degree flaw around the weld, the maximum depth of 58 percent through-wall is allowed in accordance with the design requirements of Section III of the ASME Code. Given the minimal crack growth predicted for the flaw, the staff finds that the indication has sufficient margin during the period of the proposed alternative before the flaw would reach the maximum allowable depth.

Because the examination results show essentially no crack growth, the analytical evaluation shows low fatigue crack growth, and eddy current examinations have confirmed that the indication is not surface connected, the NRC staff concludes that there is reasonable assurance that the indication will not propagate significantly between 2013 and 2020. Therefore, the staff finds that the flaw will not challenge the structural integrity of weld RC-32-MRCL-AIII-03 during the time period of the licensee’s proposed alternative.

Based on the information submitted, the NRC staff concludes that it is acceptable to defer the required examination in the second period of the Point Beach, Unit 1 fifth 10-year ISI interval. The staff finds that the licensee adequately demonstrated that the structural integrity of weld RC-32-MRCL-AIII-03 will not be challenged, and that the proposed alternative provides an acceptable level of quality and safety.

#### 4.0 CONCLUSION

As set forth above, the NRC staff reviewed the licensee's submittal and determines that use of the proposed alternative presented in Relief Request 1-RR-6 provides an acceptable level of quality and safety.

Accordingly, the NRC staff concludes that the licensee has adequately addressed all of the regulatory requirements set forth in 10 CFR 50.55a(a)(3)(i), and is compliance with the ASME Code's requirements. Therefore, the NRC staff authorizes the one-time use of 1-RR-6 at the Point Beach Nuclear Power Plant, Unit 1, for the duration up to and including the next scheduled inservice inspection of the RPV that is currently scheduled in 2020, not to exceed the limits of the fifth 10-year inservice inspection interval which ends on July 30, 2022.

All other ASME Code, Section XI requirements for which relief was not specifically requested and approved in the subject request for relief remain applicable, including third-party review by the Authorized Nuclear Inservice Inspector.

Principal Contributor: J. Collins, NRR/DE

Date: December 10, 2014

E. McCartney

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If you have any questions, please contact Terry Beltz at (301) 415-3049, or via e-mail at Terry.Beltz@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

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Safety Evaluation

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**ADAMS Accession No.: ML14343A051**

\* via email dated December 5, 2014

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