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

July 21, 2010

Mr. Larry Meyer
Site Vice President
NextEra Energy Point Beach, LLC
6610 Nuclear Road
Two River, WI 54241

SUBJECT: POINT BEACH NUCLEAR PLANT, UNITS 1 AND 2 - EVALUATION OF RELIEF REQUEST RR-22 (TAC NOS. ME2146 AND ME2147)

Dear Mr. Meyer:

By letter dated August 28, 2009 (Agencywide Documents Access and Management System Accession No. ML092400266), NextEra Energy Point Beach, LLC (the licensee) submitted a request to the U.S. Nuclear Regulatory Commission (NRC) for relief from the requirements of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code), Section XI, 1998 Edition through 2000 Addenda, IWB-5222(b) which requires end-of-interval system leakage test examinations conducted on selected Class 1 component pressure boundaries at the Point Beach Nuclear Plant (PBNP), Units 1 and 2.

Specifically, pursuant to Title 10 of the *Code of Federal Regulations* (10 CFR), Part 50, Section 50.55a(a)(3)(ii), the licensee requested to use an alternative on the basis that complying with the specified requirement would result in hardship or unusual difficulty. The proposed alternative would pressurize up to the inboard isolation valve, which would exclude a segment of the Class 1 boundary from attaining the required test pressure. The licensee would continue to perform the visual examination of the piping segments between the inboard and the outboard isolation valves.

The NRC staff has reviewed the proposed request and concludes, as set forth in the enclosed safety evaluation, that the licensee has adequately addressed all the regulatory requirements set forth in 10 CFR 50.55a(a)(3)(ii), and is in compliance with ASME Code requirements. Therefore, the staff authorizes the licensee's proposed alternative for the remainder of the fourth 10-year inservice inspection interval at the PBNP-1 and PBNP-2 which ends on June 30, 2012.

If you have any questions, please contact Terry Beltz of my staff at (301) 415-3049.

Sincerely,

Robert J. Pascarelli, Chief
Plant Licensing Branch III-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. 50-266 and 50-301

Enclosure:
Safety Evaluation

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UNITED STATES
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SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELIEF REQUEST RR-22

ALTERNATIVE TO THE FOURTH 10-YEAR INSERVICE INSPECTION PROGRAM INTERVAL

NEXTERA ENERGY POINT BEACH, LLC

POINT BEACH NUCLEAR PLANT, UNITS 1 AND 2

DOCKET NOS. 50-266 AND 50-301

1.0 INTRODUCTION

By letter dated August 28, 2009 (Reference 1), NextEra Energy Point Beach, LLC (the licensee) submitted Relief Request RR-22 for the Point Beach Nuclear Plant (PBNP), Units 1 and 2, proposing an alternative to the requirements of American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code), Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components," 1998 Edition through 2000 Addenda, IWB-5222(b) which requires the end-of-interval system leakage test to include all ASME Code Class 1 components within the system boundary.

Specifically, pursuant to Title 10 of the *Code of Federal Regulations* (10 CFR), Part 50, Section 50.55a(a)(3)(ii), the licensee requested to use the proposed alternative on the basis that complying with the specified requirement would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety. The licensee proposed an alternative to pressurize up to the inboard isolation valve, which would exclude a segment of the Class 1 boundary from attaining the required test pressure. The licensee would continue to perform the visual examination of the piping segments between the inboard and the outboard isolation valves. The licensee's request for relief is based on the hardship of performing off-normal activities in order to pressurize the portion of piping between the inboard and outboard isolation valves to Code Class 1 system leakage test pressure corresponding to 100 percent rated reactor power.

2.0 REGULATORY REQUIREMENTS

The regulations in 10 CFR 50.55a(g) require that inservice inspection (ISI) of ASME Code Class 1, 2, and 3 components be performed in accordance with Section XI of the ASME Code and applicable addenda, except where specific written relief has been granted by the Commission pursuant to 10 CFR 50.55a(g)(6)(i). According to 10 CFR 50.55a(a)(3), alternatives to the

Enclosure

requirements of paragraph 50.55a(g) may be used, when authorized by the Nuclear Regulatory Commission (NRC), if an applicant demonstrates that the proposed alternatives would provide an acceptable level of quality and safety or if the specified requirement would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety.

Pursuant to 10 CFR 50.55a(g)(4), ASME Code Class 1, 2, and 3 components (including supports) shall meet the requirements, except the design and access provisions and the preservice 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 design, geometry, and materials of construction of the components. The regulations require that ISI 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 ISI Code of Record for the fourth 10-year inspection interval of PBNP-1 and PBNP-2 is the 1998 Edition through 2000 Addenda of the ASME Code, Section XI.

3.0 TECHNICAL EVALUATION

3.1 Components for Which Relief is Requested

The ASME Code components affected by this relief are associated with the Reactor Coolant Pressure Boundary (RCPB) and include the following:

- Reactor Coolant System (RCS) Loop and Pressurizer Sample Double Valve Isolation Segments
- Vents, Drain, Instrumentation and Test Connection Double Valve Isolation Segments

3.2 ASME Code Requirements for Which Relief is Requested

Relief is requested from the 1998 Edition including addenda up to the 2000 of the ASME Code, Section XI, Paragraph IWB-5222(b) in Examination Category B-P of Items B15.50 and B15.70 that require that the pressure retaining boundary during the system leakage test be conducted at or near the end of each inspection interval extend to all Class 1 pressure retaining components within the system boundary.

3.3 Licensee Request for Relief

The licensee is requesting relief from performing the system leakage test in accordance with the requirements of the 1998 Edition of the ASME Code, Section XI through the 2000 Addenda, Paragraph IWB-5222(b) for the portion of Class 1 piping segments between the inboard and the outboard isolation valves identified in Section 3.1 and listed in Attachment 1 of Reference 1.

3.4 Licensee's Proposed Alternative

The licensee proposes that during the remainder of the fourth 10-year ISI program interval, system leakage tests on Class 1 pressure retaining components within the system boundary be performed with the inboard and outboard isolation valves configured in their normal reactor startup position. The VT-2 visual examination for leakage will extend to and include the second closed isolation valve or closure device at the boundary extremity.

3.5 Licensee Basis for the Alternative

Performing a leakage test of the Class 1 boundary beyond the inboard isolation valves at or near the end of each inspection interval requires conditions that place the plant in abnormal configurations or requires off-normal activities in order to pressurize the subject piping. These challenges include abnormal line-ups, installing jumpers around valve operation interlocks, installing and removing piping jumpers around valves, removing valve internals, and installing plugs. Associated with each challenge come additional burdens prior to plant restart, such as:

- Valve manipulations which add unnecessary challenges to maintaining the plant in a safe configuration. In some cases, the impracticality of manually opening inboard isolation valves (e.g., check valves) mandates alternate lineups that challenge system integrity.

- System preparations and restorations required inside containment including radiological restricted areas that increase radiological exposure to plant personnel, contaminate test equipment and create avoidable radiological waste.

- Routing temporary hoses/piping containing high pressure RCS fluid throughout containment, thereby creating significant personnel safety and radiological exposure hazards. The risks are further compounded by the tripping hazards plant workers inside containment must endure as a result of the hoses being routed throughout.

- Reliance upon a single closure device past the first isolation valve to contain RCS pressure from lower design pressure components and piping. This creates a significant personnel safety hazard and could lead to permanent damage to plant equipment. In addition, maintaining the requisite boron concentration in the RCS could be challenged.

These off-normal configurations and challenges may also contribute to the burden of delaying normal plant start-up because of the critical path time and effort required to ensure system configuration is restored and tested.

The licensee believes that subjecting the applicable piping segments to RCS pressure is not necessary to adequately conduct the Code-required VT-2 visual examinations for the detection of leakage or evidence of past leakage. The proposed alternative method maintains RCS barriers intact during the VT-2 visual examinations, rather than opening or bypassing the first isolation barrier prior to the examination. The Class 1 piping between the inboard and the outboard isolation valves is normally pressurized, albeit at a lower pressure, by stabilized pressure from normal seat leakage originating at the first isolation valve. This pressure is sufficient for detecting leakage and/or evidence of past leakage during system pressure tests. Therefore, the licensee proposes to validate and document the pressure boundary integrity of these piping segments and components using identical VT-2 visual examination requirements during reactor start-up following each refueling outage. This alternative would result in saving significant personnel exposure and minimizing the risk of personnel injury or contamination associated with opening or bypassing normally closed isolation devices.

3.6 NRC Staff Evaluation

The ASME Code, Section XI of Record requires that all Class 1 components within the RCS boundary undergo a system leakage test at or near the end of each inspection interval. In RR-22, the licensee proposed an alternative to test the Class 1 piping segments between the inboard and outboard isolation valves including the isolation valves in the RCPB identified in Section 3.1 and listed in Attachment 1 of Reference 1. The licensee's proposed alternative is to configure the inboard/outboard isolation valves or closure devices in their normal reactor startup position during system leakage tests on Class 1 pressure retaining components and perform the VT-2 visual examination for leakage of the piping segment extending to and including the second closed isolation valve or closure device at the boundary extremity.

The nominal operating pressure for the components is that of its connecting system unless the inboard check valve leaks. In order to perform the Code-required system leakage test for these components in the extended Class 1 pressure boundary, an alternative method of pressurizing it to the RCS operating pressure corresponding to 100 percent power would be required. The staff believes that the provision for pressurization for the system leakage test would require considerable man-hour effort resulting in high radiological exposure to personnel. Furthermore, pressurization by this method would preclude the RCS double valve isolation and may cause safety concerns for the personnel performing the examination.

The Class 1 piping between the inboard and the outboard isolation valves is normally pressurized, albeit at a lower pressure, by stabilized pressure from normal seat leakage originating at the first isolation valve. This pressure is sufficient for detecting leakage and/or evidence of past leakage during system pressure tests. This alternative, however, would expose the extended Class 1 boundary to a lower test pressure that corresponds to the operating pressure of each connecting system in lieu of the Code-required RCS pressure corresponding to 100 percent power. The staff believes that the lower pressure system leakage test of the components in the extended Class 1 boundary will also detect any leakage in the pressure boundary at a lower leak rate than that of the Code-required test pressure. Nevertheless, the components in the extended Class 1 boundary are exposed to a lower pressure than the RCS pressure during normal operation or during accident condition. Additionally, if the inboard check valve would leak (thereby pressurizing the subject components) with a through-wall flaw existing in the subject component that could only be detected at the higher pressure than that of the normal operating pressure, the flaw would be detected during a routine system leakage test of the RCS conducted prior to startup of the unit following each refueling outage. A mitigating factor in accepting the test pressure of system operating pressure in lieu of the Code-required test pressure is based on the fact that there is no known degradation mechanism, such as intergranular stress-corrosion cracking, primary water stress-corrosion cracking, or thermal fatigue, that is likely to affect the welds in the subject segments.

The staff believes that the licensee's proposed alternative provides reasonable assurance of structural integrity for the components in the extended Class 1 boundary while maintaining personnel radiation exposure to as low as reasonably achievable. The staff has further determined that compliance to the Code requirement would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety.

4.0 CONCLUSION

As set forth above, the NRC staff concludes that the proposed alternative described in relief request RR-22 demonstrates that the specified requirements would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety. The proposed alternative provides reasonable assurance of structural integrity or leak tightness of the specified components in the extended Class 1 pressure boundary. Therefore, the NRC authorizes the proposed alternative in accordance with 10 CFR 50.55a(a)(3)(ii) for the remainder of fourth 10-year ISI interval at PBNP-1 and PBNP-2.

All other ASME Code, Section XI requirements for which relief was not specifically requested and approved in relief request RR-22 remain applicable, including a third-party review by the Authorized Nuclear Inservice Inspector.

5.0 REFERENCES

1. Letter from Mr. Larry Meyer, Site Vice President, to the U.S. Nuclear Regulatory Commission, Point Beach Nuclear Plant, Units 1 and 2, Dockets 50-266 and 50-301, Renewed License Nos. DPR-24 and DPR-27, "10 CFR 50.55a Request, Relief Request RR-22, System Leakage Test – Boundaries, Fourth Ten-Year Inservice Inspection Program Interval," dated August 28, 2009 (ADAMS Accession No. ML092400266).

Principal Contributor: Prakash Patniak, NRR/DCI/CSGB

Date: July 21, 2010

If you have any questions, please contact Terry Beltz of my staff at (301) 415-3049.

Sincerely,

/RA/

Robert J. Pascarelli, Chief
Plant Licensing Branch III-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. 50-266 and 301

Enclosure:
Safety Evaluation

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ADAMS Accession No.: ML101970032 (*) memo dated 04/26/2010

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