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

December 13, 2019

Mr. Don Moul
Vice President, Nuclear Division and
Chief Nuclear Officer
Florida Power & Light Company
NextEra Energy Seabrook, LLC
Mail Stop: NT3/JW
15430 Endeavor Drive
Jupiter, FL 33478

SUBJECT: SEABROOK STATION, UNIT NO. 1 – RELIEF FROM THE REQUIREMENTS OF THE AMERICAN SOCIETY OF MECHANICAL ENGINEERS BOILER AND PRESSURE VESSEL CODE (EPID L-2019-LLR-0004)

Dear Mr. Moul:

By letter dated January 11, 2019 (Agencywide Documents Access and Management System Accession No. ML19011A331), NextEra Energy Seabrook, LLC (the licensee) submitted Relief Request 3IR-17 to the U.S. Nuclear Regulatory Commission (NRC) for relief from the requirements of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (Code), Section XI, to extend the inservice inspection (ISI) interval for Category B-A and B-D examinations at Seabrook Station, Unit No. 1 (Seabrook).

Specifically, pursuant to Title 10 of the *Code of Federal Regulations* (10 CFR) 50.55a(z)(1), the licensee requested to extend the interval for ASME Code Table IWB-2500-1, Category B-A and B-D component examinations. The licensee requested to extend the inspection interval from August 18, 2020, in the third 10-year ISI interval to no later than 2029 in the fourth 10-year ISI interval on the basis that the proposed alternative provides 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 the licensee has demonstrated that the proposed alternative 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(z)(1). Therefore, the NRC staff authorizes the use of Relief Request 3IR-17 at Seabrook for the extended third ISI interval for ASME Category B-A and B-D items until 2029.

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

If you have any questions, please contact the Seabrook Project Manager, Justin Poole, at 301-415-2048 or by e-mail to Justin.Poole@nrc.gov.

Sincerely,

James G. Danna, Chief
Plant Licensing Branch I
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-443

Enclosure:
Safety Evaluation

cc: Listserv

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SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELIEF REQUEST 3IR-17 REGARDING

THIRD 10-YEAR INSERVICE INSPECTION PROGRAM INTERVAL

NEXTERA ENERGY SEABROOK, LLC, ET AL.*

SEABROOK STATION, UNIT NO. 1

DOCKET NO. 50-443

1.0 INTRODUCTION

By letter dated January 11, 2019 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML19011A331), NextEra Energy Seabrook, LLC (the licensee) submitted Relief Request 3IR-17 to the U.S. Nuclear Regulatory Commission (NRC) for relief from the requirements of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (Code), Section XI, to extend the inservice inspection (ISI) interval for Category B-A and B-D examinations at Seabrook Station, Unit No. 1 (Seabrook).

Specifically, pursuant to Title 10 of the *Code of Federal Regulations* (10 CFR) 50.55a(z)(1), the licensee requested to extend the interval for ASME Code Table IWB-2500-1, Category B-A and B-D component examinations. The licensee requested to extend the inspection interval from August 18, 2020, in the third 10-year ISI interval to no later than 2029 in the fourth 10-year ISI interval on the basis that the proposed alternative provides an acceptable level of quality and safety.

2.0 REGULATORY EVALUATION

Adherence to Section XI of the ASME Code is mandated by 10 CFR 50.55a(g)(4), "Inservice inspection standards requirements for operating plants," which states, in part, that ASME Code Class 1, 2, and 3 components will meet requirements, except design and access provisions and preservice examination requirements, set forth in Section XI of the ASME Code.

Section 50.55a(z) of 10 CFR, "Alternatives to codes and standards," states:

> Alternatives to the requirements of paragraphs (b) through (h) of [10 CFR 50.55a] or portions thereof may be used when authorized by the Director, Office of Nuclear Reactor Regulation [this authorization has been delegated to the management of the Division of Operating Reactor Licensing].... A proposed alternative must be submitted and authorized prior to implementation. The applicant or licensee must demonstrate that:
>
> (1) *Acceptable level of quality and safety.* The proposed alternative would provide an acceptable level of quality and safety; or

Enclosure

(2) *Hardship without a compensating increase in quality and safety.* Compliance with the specified requirements of this section 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 the use of an alternative and the NRC to authorize the proposed alternative.

3.0 TECHNICAL EVALUATION

3.1 Background

The NRC staff's review of this proposed alternative assesses the consistency of the licensee's proposal with WCAP-16168-NP-A, Revision 3, "Risk-Informed Extension of the Reactor Vessel In-Service Inspection Interval," dated October 2011 (ADAMS Accession No. ML11306A084) (hereafter referred to as WCAP-A). WCAP-A provides a basis for the acceptability of the proposed inspection intervals for Category B-A and B-D components at U.S. pressurized-water reactors (PWRs) designed by Westinghouse, Combustion Engineering, and Babcock & Wilcox (B&W) through the use of risk-informed analyses and probabilistic fracture mechanics for a pilot plant of each design. WCAP-A also contains the NRC staff's safety evaluation (SE) of the Westinghouse proposal. In its SE, the NRC staff found the proposal acceptable for use based on its consistency with the principles contained in Regulatory Guide (RG) 1.174, Revision 1, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," dated November 2002 (ADAMS Accession No. ML023240437). However, the SE imposes a condition that states licensees should provide plant-specific information in six areas to demonstrate the applicability of WCAP-A to the licensee's plant. The plant-specific information requested by the condition is as follows:

(1) Licensees should provide the 95th percentile total through-wall cracking frequency ($TWCF_{TOTAL}$) and the supporting material properties at the end of the proposed 20-year ISI interval. The 95th percentile $TWCF_{TOTAL}$ should be calculated using the methodology in NUREG-1874, "Recommended Screening Limits for Pressurized Thermal Shock (PTS)" (ADAMS Accession No. ML070860156), which is frequently referred as "the NRC PTS Risk Study." The RT_{MAX-X} (the material property which characterizes the reactor vessel's resistance to fracture initiating from flaws found along a specific reactor pressure vessel (RPV) material and the temperature shift in the Charpy transition temperature produced by irradiation defined at the 30 foot-pound energy level (ΔT_{30}), should be calculated using the latest revision of RG 1.99, Revision 2, "Radiation Embrittlement of Reactor Vessel Materials" (ADAMS Accession No. ML003740284) or other NRC-approved methodology.

(2) Licensees should report whether the frequency of the limiting design-basis transients during prior plant operation is less than the frequency of the design-basis transients identified in the Pressurized Water Reactor Owners Group (PWROG) fatigue analysis as significant contributors to fatigue crack growth.

(3) Licensees should report the results of prior ISI of RPV welds and the proposed schedule for the next 20-year ISI interval. Each licensee should identify the years in which future inspections will be performed, and the dates provided should be within plus or minus one refueling cycle of the dates identified in the implementation plan provided to the NRC by

PWROG letter OG-10-238, "Revision to the Revised Pan for Plant-Specific Implementation of Extended Inservice Inspection Interval per WCAP-16168-NP," Revision 1 (ADAMS Accession No. ML11153A033), dated July 12, 2010 (ADAMS Accession No. ML11153A033).

(4) Licensees with B&W plants should (a) verify that the fatigue crack growth of 12 heatup and cooldown transients per year used in the PWROG fatigue analysis bounds the fatigue crack growth for all of its design-basis transients and (b) identify the design-bases transients that contribute to significant fatigue crack growth.

(5) Licensees with RPV forgings that are susceptible to underclad cracking and with $RT_{MAX\ FO}$ values exceeding 240 degrees Fahrenheit should submit a plant-specific evaluation because the analyses performed in WCAP-A are not applicable.

(6) Licensees seeking second or additional interval extensions should provide the information and analyses requested in Section 50.61a(e) of 10 CFR, "Examination and Flaw Assessment Requirements."

3.2 ASME Code Components Affected

The affected components are the RPV welds and full penetration nozzle welds. The following examination categories and item numbers from IWB-2500 and Table IWB-2500-1 of the ASME Code, Section XI, are listed in Relief Request 3IR-17.

Exam Category	Item Number	Description
B-A	B1.10	Shell Welds
B-A	B1.11	Circumferential Shell Welds
B-A	B1.12	Longitudinal Shell Welds
B-A	B1.20	Head Welds
B-A	B1.21	Circumferential Head Welds
B-A	B1.22	Meridional Head Welds
B-A	B1.30	Shell-to-Flange Weld
B-A	B1.40	Head -to-Flange Weld
B-D	B3.90	Nozzle-to-Vessel Welds
B-D	B3.100	Nozzle Inner Radius Section

3.3 Applicable Code Edition and Addenda

The Code of record for the third ISI inspection of ASME Code Class 1, 2, and 3 components is the 2004 Edition of the ASME Code, Section XI.

3.4 Applicable Code Requirements

ASME Code, Section XI, paragraph IWB-2412, "Inspection Program B," requires volumetric examination of essentially 100 percent of the total number of RPV pressure-retaining welds identified in Table IWB-2500-1 once each 10-year interval. The current third ISI ends on August 18, 2020. The applicable Code for the fourth 10-year ISI interval will be determined in accordance with the requirements of 10 CFR 50.55a.

3.5 Licensee's Proposed Alternative

In Relief Request 3IR-17, the licensee proposed to extend the third ISI interval for the ASME Code Category B-A and B-D examination items from August 18, 2020, to the year 2029 for Seabrook. The licensee plans to perform the ASME Code-required examination of the subject items no later than 2029, which is consistent with the schedule proposed in PWROG letter (OG-10-238), dated July 12, 2010.

3.6 Licensee's Basis for Alternative

The licensee stated that the alternative is based on a negligible change in risk satisfying the risk criteria specified in RG 1.174, Revision 1. The licensee further stated that the methodology used to conduct this analysis is based on the study defined in WCAP-A. This study focuses on risk assessments of materials within the beltline region of the RPV wall. Appendix A of the WCAP-A identifies the parameters to be compared between an applicant's plant and the appropriate pilot plant. These items include:

- Dominant PTS Transients in the NRC PTS Risk Study
- Through Wall Cracking Frequency (TWCF)
- Frequency and Severity of Design Basis Transients
- Cladding Layers (Single/Multiple)

Table 1, "Critical Parameters for the Application of Bounding Analysis for Seabrook Unit 1," of Relief Request 3IR-17 provides the above parameters for Seabrook and the Westinghouse pilot plant. Based on this information, the licensee concludes that the parameters for Seabrook are bound by the results of the Westinghouse pilot plant and, thus, qualify for ISI interval extensions.

For the most important parameter (TWCF), the licensee's calculated values are 1.32E-13 events per year for Seabrook, as compared to the WCAP-A TWCF of 1.76E-08 events per year for the Westinghouse pilot plant.

Table 2 of Relief Request 3IR-17 also contains inspection showing that RPV examinations have been performed with satisfactory results.

3.7 Duration of Alternative

The relief request, if granted, would extend the third 10-year ISI interval from August 18, 2020, to the year 2029 for Seabrook. ASME Category B-A and B-D items are listed in Section 3.2 of this SE.

3.8 NRC Staff Evaluation

Since the WCAP-A methodology has already been accepted by the NRC staff, the current evaluation focused on the manner in which the licensee addressed the four critical parameters in Table A-1 of WCAP-A, Appendix A (the four critical parameters are provided in SE Section 3.6), and the six plant-specific information items specified in the NRC SE enclosed in WCAP-A (six plant-specific information items provided in SE Section 3.1). The NRC staff reviewed the licensee's evaluation of the four critical parameters and the six plant-specific information items.

Regarding PTS transients, the licensee identified the NRC letter report, "Generalization of Plant-Specific Pressurized Thermal Shock (PTS) Risk Results to Additional Plants" (ADAMS Accession No. ML042880482), revised December 14, 2004, as its plant-specific basis. This is acceptable because the SE in WCAP-A concludes that based on this letter report, the PTS transient characteristics are generally applicable for plants from the same reactor vendor. Regarding the cladding layers, the licensee stated in its submittal that Seabrook is a "single layer" unit. This is also acceptable because it is consistent with the Westinghouse pilot plant.

The remaining two critical parameters are among the six plant-specific information items discussed below.

3.8.1 Plant-Specific Information Item (1)

Plant-specific Information Item 1 addresses TWCFs, and these critical parameters are addressed in Table 3 of the licensee's submittal. As specified in the guidance in Appendix A of WCAP-A, the licensee provided Tables 3 in its submittal, which contains a summary of the input parameters for all RPV materials and the resulting TWCFs for the controlling materials. The licensee proposed that the information specified in Table 3 demonstrates that Seabrook is bounded by WCAP-A, and is, therefore, acceptable. Specifically, Table 3 of Relief Request 3IR-17 provides input chemistry data, unirradiated nil-ductility transition reference temperature (RT_{NDT}), neutron fluence values for all RPV materials, and output shift and TWCF for controlling RPV materials at Seabrook.

The NRC staff compared the Table 3 information with that in the license renewal application (LRA) for Seabrook (ADAMS Accession No. ML101320273). Because these values were accepted in the SE dated January 2, 2019, for the LRA, "Safety Evaluation Report Related to the License Renewal of Seabrook Station" (ADAMS Accession No. ML18362A370), they are considered as the current licensing basis values. The NRC staff noted the following: (1) fluence values in Table 3 of the submittal are consistent with the LRA values listed in Table 4.2.3-1, (2) the nickel and copper values for all reported welds and plates for Seabrook in the submittal are consistent with the LRA values, and (3) the chemistry factors addressed in the submittal are consistent with the LRA values. Therefore, the NRC staff accepts these values.

Fluence Values as Reported in the Submittal

The NRC staff noted that the LRA fluence values are for 55 effective full power years (EFPY) and the relief request fluence values are for 36 EFPY, as addressed in WCAP-17441-NP, Revision 0, "Seabrook Unit 1 Heatup and Cooldown Limit Curves for Normal Operation," dated October 2011 (ADAMS Accession No ML12341A096). The fluence values as reported in WCAP-17441-NP were used in developing pressure-temperature limits in the submittal (ADAMS Accession Nos. ML14216A405 and ML12341A096). The NRC staff finds this difference in values reported in the submittal for 36 EFPY to be acceptable because the value provided in the LRA is required to predict the EFPY incurred by the end of the period of extended operation, while the value used in this request needs to only predict EFPY to the end of the request (i.e.,36 EFPY).

Using linear interpolation, the NRC staff found that the highest LRA fluence value of 3.59E+19 neutrons per square centimeter (n/cm^2) for 55 EFPY for Seabrook will become 2.0E+19 n/cm^2 for 36 EFPY, which is lower than the value of 3.05E+19 n/cm^2 reported in Relief Request 3I-17 for 36 EFPY for all RPV materials of Seabrook. The value used in Relief Request 3I-17 is higher than the actual fluence value for the 36 EFPY, as addressed in WCAP-17441-NP, and therefore, the lower fluence value reflected for EFPY reported in the relief request is acceptable for 36 EFPY.

Relief Request RR-3I-17 reports the same copper and nickel values for all the reported base metal and welds for Seabrook, as addressed in the NRC staff-approved LRA. The NRC staff accepts copper and nickel values of the base metal and the welds for Seabrook in this application because they are consistent with the values used in the staff-approved LRA values.

3.8.2 Plant-Specific Information Item (2)

The NRC staff reviewed plant-specific Information Item (2) regarding the frequency of the limiting design-basis transients. The licensee stated in Table 1 of its submittal that the heatup and cooldown cycles per year are bounded by the heatup and cooldown cycles for the Westinghouse pilot plant. The NRC staff examined the heatup and cooldown design and projected cycles for 60 years of operation in Table 4.3.1-3 of the LRA for the Seabrook. The NRC staff found that the frequencies (heatup and cooldown) are below the bounding value of 7 per year for the Westinghouse pilot plant. The NRC staff noted that the bounding value was conservatively established by using conservative values for Westinghouse units. Any heatup and cooldown that might have occurred prior to the commission of Seabrook could affect the mechanical properties of the vessel. Establishing a conservative value for the design-basis transients will compensate for any effect on the RPV welds prior to commissioning the Seabrook. Therefore, the NRC staff concludes that the licensee has acceptably addressed plant-specific Information Item (2).

3.8.3 Plant-Specific Information Item (3)

The NRC staff reviewed plant-specific Information Item (3) regarding the results of prior ISIs of RPV welds and the proposed schedule for the extended ISI interval. Table 2 in the licensee's submittal provided additional information pertaining to previous RPV inspections and the schedule for the future inspection. Specifically, Table 2 indicated that two 10-year ISIs have been performed for Seabrook. There were 18 indications identified in the beltline region of the RPV during the last ISI. The subsurface indications were located in the upper, intermediate, and lower shell axial weld seams, and the intermediate to lower shell circumferential weld seam (items 10, 12, 14, and 13, respectively, in Table 3 of the submittal), and these indications were acceptable per Table IWB-3510-1 of the ASME Code, Section XI. Two of the indications are within the inner 1/10th or 1 inch of the reactor vessel thickness and required further evaluation. The NRC staff independently verified that the number of indications reported by the licensee is lower than the number of indications allowed by the criteria addressed in 10 CFR 50.61a requirements. Because both ASME Code, Section XI, and 10 CFR 50.61a requirements regarding detected flaws are met, the NRC staff determines that the licensee has adequately addressed the first part of plant-specific Information Item (3).

For the the second part of plant-specific Information Item (3), the licensee, in Relief Request RR-3I-17, proposed to conduct the next RPV inspection in 2029 for Seabrook. This year is consistent with the latest revised implementation PWROG letter (OG-10-238). Thus, the NRC staff concludes that the licensee has adequately addressed the second part of plant-specific Information Item (3). In summary, the licensee has adequately addressed plant-specific Information Item (3).

3.8.4 Plant-Specific Information Items 4 through 6

The licensee did not address plant-specific Information Items (4), (5), and (6). The NRC staff examined the specifics in each of these plant-specific information items and confirmed that these information requirements are not applicable to Seabrook. Thus, the NRC staff concludes that the plant-specific Information Items (4), (5), and (6) do not apply to Seabrook.

3.9 Summary

The NRC staff has reviewed the licensee's submittal and performed independent calculations to verify the input data and output results in Table 3 of the submittal. With this information, the NRC staff determined that the proposed alternative is based on the WCAP-A methodology, and the $TWCF_{95\text{-}TOTAL}$ values in Table 3 of the submittal are bounded by the corresponding pilot plant parameter in the WCAP-A. Consequently, the licensee has demonstrated that the proposed alternative will provide an acceptable level of quality and safety for Seabrook.

4.0 CONCLUSION

As set forth above, the NRC staff determines that the licensee has demonstrated that the proposed alternative provides an acceptable level of quality and safety. Accordingly, the NRC staff concludes that the licensee has adequately addressed all the regulatory requirements set forth in 10 CFR 50.55a(z)(1). Therefore, the NRC staff authorizes the use of Relief Request RR-3IR-17 at Seabrook for the extended third ISI interval for ASME Category B-A and B-D items until 2029.

All other requirements of the ASME Code, Section XI, for which relief has not been specifically requested remain applicable, including third party review by the Authorized Nuclear Inservice Inspector.

Principal Contributor: G. Cheruvenki

Date: December 13, 2019

SUBJECT: SEABROOK STATION, UNIT NO. 1 – RELIEF FROM THE REQUIREMENTS OF THE AMERICAN SOCIETY OF MECHANICAL ENGINEERS BOILER AND PRESSURE VESSEL CODE (EPID L-2019-LLR-0004) DATED DECEMBER 13, 2019

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