January 11, 2019

10 CFR 50.55a
Docket No. 50-443
SBK-L-19003

U.S. Nuclear Regulatory Commission
Attention: Document Control Desk
One White Flint North
11555 Rockville Pike
Rockville, MD 20582

Seabrook Station
Relief Request Proposed Alternative in Accordance with 10 CFR 50.55a(z)(1), 3IR-17
Implementation of an Extended Reactor Vessel Inservice Inspection Interval

In accordance with the provisions of 10 CFR 50.55a(z)(1), NextEra Energy Seabrook, LLC (NextEra) requests approval to extend the inservice inspection interval for the Seabrook Unit 1 Reactor Vessel (RV) from 2020 to 2029.

NextEra proposes to implement an alternative to the requirement of ASME Section XI IWB-2412, Inspection Program B, that volumetric examination of RV Examination categories B-A and B-D be performed once each 10-year ISI interval. The current third ISI interval ends on August 18, 2020. NextEra proposes to perform the third ASME Section XI Category B-A and B-D examinations in the fourth ISI interval no later than 2029. The attachment to this letter provides the basis and supporting information for the proposed alternative.

NextEra requests prompt NRC review and approval of the proposed alternative by January 15, 2020, to support the use of the proposed alternative during the refueling outage in the spring of 2020.

There are no commitments being made in this submittal.

If you have any questions regarding this submittal, please contact me at (603) 773-7932.

Sincerely,

NextEra Energy Seabrook, LLC

Kenneth J. Browne
Licensing Manager

Attachment: Proposed Alternative for Seabrook Unit 1in Accordance with 10 CFR 50.55a(z)(1)

cc:

NRC Region I Administrator
NRC Project Manager
NRC Senior Resident Inspector

Attachment to SBK-L-19003

Relief Request Number 3IR-17

<u>Proposed Alternative in Accordance with 10 CFR 50.55a(z)(1), Implementation of an Extended Reactor Vessel Inservice Inspection Interval</u>

Relief Request Number 3IR-17
Extension of Seabrook Unit 1 RPV Welds from 10 to 20 Years

Proposed Alternative
In Accordance with 10 CFR 50.55a(z)(1)

-Alternative Provides Acceptable Level of Quality and Safety-

1. ASME Code Component(s) Affected

The affected component is the Seabrook Unit 1 reactor vessel (RV), specifically, the following American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel (BPV) Code, Section XI (Reference 1) examination categories and item numbers covering examinations of the RV. These examination categories and item numbers are from IWB-2500 and Table IWB-2500-1 of the ASME BPV Code, Section XI.

Category B-A welds are defined as "Pressure Retaining Welds in Reactor Vessel."
Category B-D welds are defined as "Full Penetration Welded Nozzles in Vessels."

Examination Category	Item No.	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 Inside Radius Section

(Throughout this request the above examination categories are referred to as "the subject examinations" and the ASME BPV Code, Section XI, is referred to as "the Code.")

2. Applicable Code Edition and Addenda

ASME Code Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components," 2004 Edition (Reference 1).

3. Applicable Code Requirement

IWB-2412, Inspection Program B, requires volumetric examination of essentially 100% of reactor vessel pressure-retaining welds identified in Table IWB-2500-1 once each 10-year interval. The Seabrook Unit 1 third 10-year inservice inspection (ISI) interval is scheduled to end on August 18, 2020. The applicable Code for the fourth 10-year ISI interval will be selected in accordance with the requirements of 10 CFR 50.55a.

4. Reason for Request

An alternative is requested from the requirement of IWB-2412, Inspection Program B, that volumetric examination of essentially 100% of reactor vessel pressure-retaining Examination Category B-A and B-D welds be performed once each 10-year interval. Extension of the interval between examinations of Category B-A and B-D welds from 10 years to up to 20 years will result in a reduction in man-rem exposure and examination costs.

5. Proposed Alternative and Basis for Use

NextEra proposes not to perform the ASME Code required volumetric examination of the Seabrook Unit 1 RV full penetration pressure-retaining Examination Category B-A and B-D welds for the third inservice inspection, currently scheduled for 2020. NextEra will perform the third ASME Code required volumetric examination of the Seabrook Unit 1 RV full penetration pressure-retaining Examination Category B-A and B-D welds in the fourth inservice inspection interval in 2029. The proposed inspection date is consistent with the latest revised implementation plan in PWROG letter OG-10-238 (Reference 2).

In accordance with 10 CFR 50.55a(z)(1), an alternate inspection interval is requested on the basis that the current interval can be revised with negligible change in risk by satisfying the risk criteria specified in Regulatory Guide 1.174 (Reference 3).

The methodology used to conduct this analysis is based on that defined in the study in WCAP-16168-NP-A, Revision 3, "Risk-Informed Extension of the Reactor Vessel In-service Inspection Interval" (Reference 4). This study focuses on risk assessments of materials within the beltline region of the RV wall. The results of the calculations for Seabrook Unit 1 were compared to those obtained from the Westinghouse pilot plant evaluated in WCAP-16168-NP-A, Revision 3. Appendix A of the WCAP identifies the parameters to be compared. Demonstrating that the parameters for Seabrook Unit 1 are bounded by the results of the Westinghouse pilot plant qualifies Seabrook Unit 1 for an ISI interval extension.

Table 1 below lists the critical parameters investigated in the WCAP and compares the results of the Westinghouse pilot plant to those of Seabrook Unit 1. Tables 2 and 3 provide additional information that was requested by the NRC and included in Appendix A of Reference 4.

Table 1: Critical Parameters for the Application of Bounding Analysis for Seabrook Unit 1

Parameter	Pilot Plant Basis	Plant-Specific Basis	Additional Evaluation Required?
Dominant Pressurized Thermal Shock (PTS) Transients in the NRC PTS Risk Study are Applicable	NRC PTS Risk Study (Reference 5)	PTS Generalization Study (Reference 6)	No
Through-Wall Cracking Frequency (TWCF)	1.76E-08 Events per year (Reference 4)	1.32E-13 Events per year (Calculated per Reference 4)	No
Frequency and Severity of Design Basis Transients	7 heatup/cooldown cycles per year (Reference 4)	Bounded by 7 heatup/cooldown cycles per year	No
Cladding Layers (Single/Multiple)	Single Layer (Reference 4)	Single Layer	No

Table 2 below provides a summary of the latest reactor vessel inspection for Seabrook Unit 1 and an evaluation of the recorded indications. This information confirms that satisfactory examinations have been performed on the Seabrook Unit 1 reactor vessel.

<table>
<tr><th colspan="2">Table 2: Additional Information Pertaining to Reactor Vessel Inspection for Seabrook Unit 1</th></tr>
<tr><td>Inspection methodology:</td><td>The latest ISI for Seabrook Unit 1 was conducted in accordance with the ASME Code, Section XI, 1995 Edition, with 1996 Addenda. Examinations of Category B-A and B-D welds were performed to ASME Section XI, Appendix VIII, 1995 Edition with 1996 Addenda as modified by 10 CFR 50.55a(b)(2)(xiv, xv, and xvi). Future inservice inspections will continue to be performed to ASME Section XI, Appendix VIII methodology.</td></tr>
<tr><td>Number of past inspections:</td><td>Two 10-year inservice inspections and a preservice inspection have been performed.</td></tr>
<tr><td>Number of indications found:</td><td>There were 18 indications identified in the beltline region of the RV during the last ISI. The subsurface indications are 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), and are acceptable per Table IWB-3510-1 of Section XI of the ASME Code. Two of these indications are within the inner 1/10th or 1 inch of the reactor vessel thickness and required further evaluation. These indications are acceptable per the requirements of the Alternate PTS Rule, 10 CFR 50.61a (Reference 7).

A disposition of the two flaws against the limits of the Alternate PTS Rule is shown in the table below. Both flaws are located within the plate material of the reactor vessel.

<table>
<tr><th colspan="2">Through-Wall Extent, TWE (in)</th><th rowspan="2">Scaled Maximum number of flaws per 8132 square-inches of inside surface area in the inspection volume that are greater than or equal to TWE_{MIN} and less than TWE_{MAX}. This flaw density does not include underclad cracks in forgings.</th><th rowspan="2">Number of Seabrook Flaws Evaluated (Axial/Circ.)</th></tr>
<tr><th>TWE_{MIN}</th><th>TWE_{MAX}</th></tr>
<tr><td>0</td><td>0.075</td><td>No Limit</td><td>0</td></tr>
<tr><td>0.075</td><td>0.375</td><td>66</td><td>2 (1/1)</td></tr>
<tr><td>0.125</td><td>0.375</td><td>26</td><td>2 (1/1)</td></tr>
<tr><td>0.175</td><td>0.375</td><td>7</td><td>1 (0/1)</td></tr>
<tr><td>0.225</td><td>0.375</td><td>3</td><td>1 (0/1)</td></tr>
<tr><td>0.275</td><td>0.375</td><td>1</td><td>1 (0/1)</td></tr>
<tr><td>0.325</td><td>0.375</td><td>1</td><td>1 (0/1)</td></tr>
<tr><td>0.375</td><td>Infinite</td><td>0</td><td>0</td></tr>
</table></td></tr>
<tr><td>Proposed inspection schedule for balance of plant life:</td><td>The third inservice inspection is currently scheduled for 2020. This inspection is proposed to be performed during the 2029 refueling outage. The proposed inspection date is consistent with the latest revised implementation plan, OG-10-238 (Reference 2).</td></tr>
</table>

Table 3 below summarizes the inputs and outputs for the calculation of through-wall cracking frequency (TWCF).

Table 3: Details of TWCF Calculation for Seabrook Unit 1 at 55 Effective Full Power Years (EFPY)									
Inputs (Reference 9)[1]									
					Upper Shell T_{wall} [inches]:				10.91
					Intermediate and Lower Shell T_{wall} [inches]:				8.79
No.	Region and Component Description	Material ID	Material Heat No.	Cu [wt%]	Ni [wt%]	R.G. 1.99 Pos.	CF [°F]	$RT_{NDT(U)}$ [°F]	Fluence [n/cm^2, E > 1.0 MeV]
1	Upper Shell (US) Plate	R1807-3, 122-102A	C4233-2	0.06	0.67	1.1	37	10	7.35E+17
2	US Plate	R1807-1, 122-102B	C4049-1	0.08	0.60	1.1	51	30	7.35E+17
3	US Plate	R1807-2, 122-102C	C4049-2	0.09	0.61	1.1	58	30	7.35E+17
4	Intermediate Shell (IS) Plate	R1806-1, 124-102A	C4036-2	0.045	0.61	1.1	28.5	40	3.05E+19
5	IS Plate	R1806-3, 124-102B	C4197-1	0.075	0.63	1.1	47.5	10	3.05E+19
6	IS Plate	R1806-2, 124-102C	A2749-2	0.06	0.64	1.1	37	0	3.05E+19
7	Lower Shell (LS) Plate	R1808-3, 142-102A	D1136-2	0.07	0.59	2.1	45.0	40	3.05E+19
8	LS Plate	R1808-1, 142-102B	D1081-3	0.06	0.58	1.1	37	40	3.05E+19
9	LS Plate	R1808-2, 142-102C	D1081-2	0.06	0.58	1.1	37	10	3.05E+19
10	US Longitudinal (Long) Welds	101-122 A,B,C	86998	0.05	0.11	1.1	38.7	-10	7.12E+17
11	US to IS Circumferential (Circ) Weld	103-121	90128	0.045	0.06	1.1	31.3	-56	7.35E+17
12	IS Long Welds	101-124 A,B,C	4P6052	0.047	0.049	1.1	30.7	-60	1.78E+19
13	IS to LS Circ Weld	101-171	4P6052	0.047	0.049	1.1	30.7	-60	3.03E+19
14	LS Long Welds	101-142 A,B,C	4P6052	0.047	0.049	1.1	30.7	-60	2.93E+19
Outputs									
Methodology Used to Calculate ΔT_{30}: Regulatory Guide 1.99, Revision 2 (Reference 8)									
	Controlling Material Region No.		α_{XX}	RT_{MAX-XX} [°R]		Fluence [n/cm^2, E >1.0 MeV]	FF (Fluence Factor)	ΔT_{30} [°F]	$TWCF_{95-XX}$
Limiting Axial Weld - AW	7		2.5000	557.50		2.93E+19	1.2851	57.83	0.000E+00
Limiting Plate - PL	7		2.5000	557.93		3.05E+19	1.2946	58.26	5.261E-14
Limiting Circumferential Weld - CW	7		2.5000	557.86		3.03E+19	1.2931	58.19	0.000E+00
								$TWCF_{95-TOTAL} = (\alpha_{AW}TWCF_{95-AW} + \alpha_{PL}TWCF_{95-PL} + \alpha_{CW}TWCF_{95-CW})$:	**1.32E-13**

(1) Material properties and fluence values for the upper, intermediate and lower shell plate and weld materials are based on WCAP-17441-NP (Reference 9).

6. Duration of Proposed Alternative

This request is applicable to the Seabrook Unit 1 inservice inspection program for the third and fourth 10-year inspection intervals.

7. Precedents

- "Surry Power Station Units 1 and 2 – Relief Implementing Extended Reactor Vessel Inspection Interval (TAC Nos. ME8573 and ME8574)," dated April 30, 2013, ADAMS Accession Number ML13106A140.

- "Vogtle Electric Generating Plant, Units 1 and 2 – Request for Alternatives VEGP-ISI-ALT-05 and VEGP-ISI-ALT-06 (TAC Nos. MF2596 and MF2597)," dated March 20, 2014, ADAMS Accession Number ML14030A570.

- "Catawba Nuclear Station Units 1 and 2: Proposed Relief Request 13-CN-003, Request for Alternative to the Requirement of IWB-2500, Table IWB-2500-1, Category B-A and Category B-D for Reactor Pressure Vessel Welds (TAC Nos. MF1922 and MF1923)," dated March 26, 2014, ADAMS Accession Number ML14079A546.

- "Sequoyah Nuclear Plant, Units 1 and 2 – Requests for Alternatives 13-ISI-1 and 13-ISI-2 to Extend the Reactor Vessel Weld Inservice Inspection Interval (TAC Nos. MF2900 and MF2901)," dated August 1, 2014, ADAMS Accession Number ML14188B920.

- "Byron Station, Unit No. 1 – Relief from Requirements of the ASME Code to Extend the Reactor Vessel Inservice Inspection Interval (TAC No. MF3596)," dated December 10, 2014, ADAMS Accession Number ML14303A506.

- "Wolf Creek Generating Station – Request for Relief Nos. I3R-08 and I3R-09 for the Third 10-Year Inservice Inspection Program Interval (TAC Nos. MF3321 and MF3322)," dated December 10, 2014, ADAMS Accession Number ML14321A864.

- "Callaway Plant, Unit 1 – Request for Relief I3R-17, Alternative to ASME Code Requirements Which Extends the Reactor Vessel Inspection Interval from 10 to 20 Years (TAC No. MF3876)," dated February 10, 2015, ADAMS Accession Number ML15035A148.

- "Braidwood Station, Units 1 and 2 – Request for Relief from the Requirements of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code) (CAC Nos. MF8191 and MF8192)," dated March 15, 2017, ADAMS Accession Number ML17054C255.

- "South Texas Project, Units 1 and 2 – Relief from the Requirements of the ASME Code Regarding the Third 10-Year Inservice Inspection Program Interval (EPID L-2018-LLR-0010)," dated July 24, 2018, ADAMS Accession Number ML18177A425.

8. References

1. ASME Boiler and Pressure Vessel Code, Section XI, 2004 Edition, American Society of Mechanical Engineers, New York.
2. PWROG Letter OG-10-238, "Revision to the Revised Plan for Plant Specific Implementation of Extended Inservice Inspection Interval per WCAP-16168-NP, Revision 1, "Risk-Informed Extension of the Reactor Vessel In-Service Inspection Interval." PA-MSC-0120," July 12, 2010 (ADAMS Accession Number ML11153A033).
3. NRC Regulatory Guide 1.174, Revision 1, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," U.S. Nuclear Regulatory Commission, November 2002, (ADAMS Accession Number ML003740133).
4. Westinghouse Report, WCAP-16168-NP-A, Revision 3, "Risk-Informed Extension of the Reactor Vessel In-service Inspection Interval," October 2011 (ADAMS Accession Number ML113060207).
5. NUREG-1874, "Recommended Screening Limits for Pressurized Thermal Shock (PTS)," U.S. Nuclear Regulatory Commission, March 2010, (ADAMS Accession No. ML15222A848).
6. NRC Letter Report, "Generalization of Plant-Specific Pressurized Thermal Shock (PTS) Risk Results to Additional Plants," U.S. Nuclear Regulatory Commission, December 14, 2004 (ADAMS Accession Number ML042880482).
7. Code of Federal Regulations, 10 CFR Part 50.61a, "Alternate Fracture Toughness Requirements for Protection Against Pressurized Thermal Shock Events," U.S. Nuclear Regulatory Commission, Washington D. C., Federal Register, Volume 75, No. 1, dated January 4, 2010 and No. 22 with corrections to part (g) dated February 3, 2010, March 8, 2010, and November 26, 2010.
8. NRC Regulatory Guide 1.99, Revision 2, "Radiation Embrittlement of Reactor Vessel Materials," U.S. Nuclear Regulatory Commission, May 1988, (ADAMS Accession No. ML003740284).
9. Westinghouse Report, WCAP-17441-NP, Revision 0, "Seabrook Unit 1 Heatup and Cooldown Limit Curves for Normal Operation," October 2011, (ADAMS Accession No. ML12341A096).