March 6, 2012

SBK-L-12040
Docket No. 50-443

U.S. Nuclear Regulatory Commission
Attention: Document Control Desk
One White Flint North
11555 Rockville Pike
Rockville, MD 20852

Seabrook Station
Response to Request for Additional Information
2011 Steam Generator Tube Inspections

References:

1. NextEra Energy Seabrook, LLC letter SBK-L-11193, "Steam Generator Tube Inspection Report, dated September 19, 2011." (ML 11266A008)
2. NRC Letter "Seabrook Station Unit 1 - Request for Additional Information Regarding the 2011 Steam Generator Tube Inspections, dated February 8, 2012." (ML 120320381)

In Reference 1, NextEra Energy Seabrook, LLC (NextEra) submitted its 2011 Steam Generator Tube Inspection Report in accordance with Seabrook Station Technical Specification 6.8.1.7, "Steam Generator Tube Inspection Report."

In Reference 2, the NRC requested additional information in order to complete its review of the report.

The attachment to this letter contains the requested additional information.

Should you have questions of a technical nature regarding the additional information, please contact Mr. Russell Lieder, Nuclear Staff Engineer, at (603) 773-7105.

A001
NRR

Should you have any questions regarding this submittal, please contact Mr. Michael O'Keefe, Licensing Manager, at (603) 773-7745.

Sincerely,

NextEra Energy Seabrook, LLC

Paul Freeman
Site Vice President

Attachment: Response to Request for Additional Information - 2011 Steam Generator Tube Inspections

cc:

W.M. Dean,	NRC Region I Administrator
J. G. Lamb,	NRC Project Manager, Project Directorate I-2
W. J. Raymond,	NRC Resident Inspector

Attachment

Seabrook Station
Response to Request for Additional Information
2011 Steam Generator Tube Inspections

Seabrook Station
Response to Request for Additional Information
2011 Steam Generator Tube Inspections

By letter dated September 19, 2011, (Agencywide Documents Access and Management System Accession Number ML 11266A008), NextEra Energy Seabrook, LLC (the licensee) submitted information summarizing the results of the 2011 steam generator (SG) tube inspections at Seabrook Station, Unit 1 (Seabrook). These inspections were performed during refueling outage (RFO) 14. In order to complete its review, the U.S. Nuclear Regulatory Commission (NRC) staff requests the following additional information:

NRC Question 1. Page 1, Section 1 - Please clarify whether the indication of axial outside diameter stress-corrosion cracking detected in 2009 was both in and below the expansion transition region, or just below the expansion transition region.

NextEra Response A study performed of the location of the bottom of the expansion transition (BET) located the R27C61 BET at 0.21 inch below the top of the tubesheet. The highest OD tensile stress is located at the bottom of the expansion transition, at the start of the transition of the tube from its expanded diameter to its nominal diameter. The crack was located about 0.26 inches below the top of the tubesheet. From the crack sizing analysis, the crack was approximately 0.10 inch in length. Therefore the crack extends from 0.21 inch to 0.31 inch below the top of the tubesheet and starts at the BET.

Probe	Call	Vpp	Phase Angle	Crack Length
+Point coil	SAI @ TSH-0.26″	0.44V	46°	0.12 inch

NRC Question 2. Page 1, Section 2 -Please clarify whether the 100 percent inspection of previously reported wear indications included all wear indications (and these were examined from 3-inches above the wear scar to 3-inches below the wear scar) or whether it just included previously reported wear indications located from 3-inches above the top of the tubesheet to 3-inches below the top of the tubesheet.

NextEra Response The 100 percent of previously reported wear indications are only those indications within the 3-inches above the tubesheet to 3-inches below the top of tubesheet plus point inspection scope.

NRC Question 3. Page 4 -Please clarify the difference between the codes NR and INR referenced in Table 2, which presumably stand for "not reportable" and "indication not reportable," respectively.

NextEra Response In table 2 for OR11, there were no PLP's reported for SG B R3/C104; SG C R40/C92; SG D R41/C27 and R42/27. The NR represents "Not Reported" and is only inserted to make the table complete. In OR13, a PLP was reported in SG B R3/C104. When this tube was tested in OR14, no PLP was found. Therefore the three letter code "INR" (indication not reportable) was used in the data base.

NRC Question 4. Page 4, Table 2 - Please discuss whether all locations where potential loose part signals were detected were inspected visually to confirm the presence of a loose part. Please clarify when the NOD [no detectable degradation] and INR [indication not reportable] codes are used for dispositioning a location us PLP [possible loose part] call was made (SG B, row 3, column 104 and SG C, row 40, column 92).

NextEra Response In the OR14 inspection, PLP's were only identified in SG A and D. There were three PLP's in SG A and two PLP's in SG D. All five PLP's were visually verified from the secondary side as having no foreign object present.

In SG B and C, no PLP's were identified by eddy current testing during the OR14 inspection, therefore no visual verification was performed for those two locations.

There is however, inconsistency in the data base as the staff has pointed out, in that in SG B, the call is INR and in SG C the call is NDD. Both calls should have been INR, since only a plus point rotating probe was being used for the steam generator inspection program. A NDD call is used when a bobbin probe is being employed.

NRC Question 5. Page 5, Section 7 -Please clarify what is meant by the statement that outside diameter and primary water SCC are not of concern for operation until RFO 15.

NextEra Response Seabrook has only detected outside diameter stress corrosion cracking. The last reported outside diameter crack indication was reported in OR13 in SG C as a single axial crack indication at the top of tube sheet. In OR14, an inspection for top of tubesheet stress corrosion cracking was performed. Neither outside diameter nor primary water stress corrosion cracking was detected.

Outside diameter stress corrosion cracking is an existing damage mechanism, whereas primary water stress corrosion cracking is a potential damage mechanism. Seabrook will examine the steam generator tubes in OR15 for both these stress corrosion cracking mechanisms. It is not expected that either the structural or leakage performance criteria would be exceeded at the next steam generator tube inspection in OR15.

NRC Question 6. Page 6 - Please confirm that the foreign object referenced in the last sentence is the same metal nail that is referenced on page 3. If it is not, please clarify the history of the foreign object on page 6.

NextEra Response The foreign object stated in the last sentence on page 6 is the same metal nail that is discussed on page 3 (Foreign Objects/PLP, third paragraph).