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

October 3, 2016

Mr. Eric McCartney
Site Vice President
Seabrook Station
NextEra Energy
626 Lafayette Rd.
Seabrook, NH 03874

SUBJECT: SEABROOK STATION, UNIT NO. 1 – REQUEST FOR ADDITIONAL INFORMATION RE: REQUEST TO EXTEND CONTAINMENT LEAKAGE TEST FREQUENCY (CAC NO. MF7565)

Dear Mr. McCartney:

By letter dated March 31, 2016, as supplemented by letter dated May 31, 2016 (Agencywide Documents Access and Management System (ADAMS) Accession Nos. ML16095A278 and ML16159A194, respectively), NextEra Energy Seabrook, LLC submitted a license amendment request to revise Technical Specification 6.15, "Containment Leakage Rate Testing Program," to require a program that is in accordance with Nuclear Energy Institute (NEI) Topical Report NEI 94-01, Revision 3-A, "Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J" (ADAMS Accession No. ML12221A202).

The U.S. Nuclear Regulatory Commission staff has determined that additional information is necessary to complete its review. The request for additional information is enclosed. The licensee agreed to provide answers to the request for additional information October 28, 2016.

If you have questions, please contact me at 301-415-2048 or by e-mail at Justin.Poole@nrc.gov.

Sincerely,

Justin C. Poole, Project Manager
Plant Licensing Branch I-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-443

Enclosure:
Request for Additional Information

cc w/enclosure: Distribution via Listserv

REQUEST FOR ADDITIONAL INFORMATION

LICENSE AMENDMENT REQUEST REGARDING

REVISION OF TS 6.15 TO ADOPT NEI 94-01, REVISION 3-A

NEXTERA ENERGY SEABROOK, LLC

SEABROOK STATION, UNIT NO. 1

DOCKET NO. 50-443

By letter dated March 31, 2016 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML16095A278), as supplemented by letter dated May 31, 2016 (ADAMS Accession No. ML16159A194), NextEra Energy Seabrook, LLC (NextEra) requested an amendment to the Seabrook Station, Unit No. 1 (Seabrook) Technical Specifications (TSs). The proposed amendment will revise Seabrook TS Section 6.15, "Containment Leakage Rate Testing Program," to allow extension of the Type A test interval up to one test in 15 years and extension of the Type C test interval up to 75 months. Responses to the request for additional information (RAI) questions listed below are needed to support the U.S. Nuclear Regulatory Commission (NRC) staff's continued technical review of the proposed license amendment request (LAR).

Probabilistic Risk Assessment Licensing Branch (APLA)

APLA-RAI-1

Section 4.2.2 of Electric Power Research Institute (EPRI) Report No. 1009325, Revision 2-A, states that, "The most relevant plant-specific information should be used to develop population dose information. The order of preference shall be plant-specific best estimate, Severe Accident Mitigation Alternative (SAMA) for license renewal, and scaling of a reference plant population dose."

Accordingly, the NRC staff reviewed results documented in NUREG-1437, Supplement 46, Volume 2, "Generic Environmental Impact Statement for License Renewal of Nuclear Plants Regarding Seabrook Station" (ADAMS Accession No. ML15209A870). Appendix F of NUREG-1437 states that the Seabrook core damage frequency (CDF) is approximately 1.2E-5 per year for both internal and external events, with the internal events, internal flooding, and external flooding CDF totaling approximately 7.8E-6 per year. Table 4-1 of Attachment 4 of the LAR cites the internal events, internal flooding, and external flooding CDF as 6.64E-6 per year (3.76E-6 + 2.86E-6 + 2.09E-8 per year). Explain the difference, justifying why a lower value is now cited. Note any changes to Tables 4-2 and 4-3 that would result if the NUREG-1437 value was used instead.

APLA-RAI-2

Notes (4) and (5) in Section 5.2 of Attachment 4 of the LAR indicate that Accident Class 7 and Class 8 plant-specific person-Roentgen equivalent man (rem) doses are assigned as the frequency-weighted average of the dose from pertinent release categories (LE4, LL3, LL4, LL5, and SELL for Class 7 and SE1, SE2, LE1, and LE2, for Class 8). The NRC staff performed a confirmatory calculation using the frequencies and population doses provided in Tables 4-2 and 4-4 of Attachment 4 of the LAR, respectively (analogous to Notes (1) and (2) in Section 5.2 of Attachment 4), and calculated values of 9.83E6 and 2.63E6 person-rem (50 miles). These values reflect increases of 1.8 and 1.3 person-rem, or approximately 6 percent and 311 percent, in the total person-rem per year compared to those reported in Table 5-5, for Class 7 and Class 8, respectively.

Provide an explanation for this apparent discrepancy and/or correct the calculations for the Class 7 and Class 8 population dose estimates and any subsequent calculations and tables based on this result.

APLA-RAI-3

Section 6-3 of Attachment 4 of the LAR indicates that the large early release frequency (LERF) contribution from fire events (1.3E-10) appears to be atypically low (approximately four orders of magnitude) when compared to the corresponding fire CDF, especially given that the LERF from the total internal and external events is approximately only two orders of magnitude lower than its corresponding CDF. From Table 4-2, there are four release categories that contribute to LERF (LE1 – LE4). Of these, at least categories LE2 (containment bypass via interfacing loss-of-coolant accident (ISLOCA) through residual heat removal pipe rupture (unscrubbed release)), LE3 (containment isolation failure (large penetration, containment overpressure values)), and LE4 (long-term containment basement failure with delayed evacuation), could plausibly result from fire initiators. For example, the following scenarios are plausible: fire-induced opening or failure to close of containment isolation valves in piping, fire-induced opening or failure to close of containment overpressure valves, and fire-induced transients that lead to a LOCA via a stuck-open pressure operated relieve valve(s) and/or failure to close corresponding block valves.

Table 4-2 indicates that approximately 70 percent of the LERF contribution arises from these three release categories. Therefore, even if only 10 percent of their non-fire LERF were attributed to fire, the corresponding fire LERF-to-CDF ratio would be approximately two orders of magnitude of the fire CDF ((0.1)(1.81E-8 + 8.59E-10 + 9.20E-8)/(1.48E-6) = 0.0075), consistent with the cited total LERF-to-CDF ratio. Notably, for seismic events, the LERF-to-CDF ratio is also approximately two orders of magnitude (9.85E-8/3.25E-6 = 0.030).

Provide additional explanation for why the fire events contribution to LERF is approximately four, rather than approximately two, orders of magnitude less than the fire CDF. If this is incorrect, provide the revised value for fire LERF and discuss any changes to the overall analysis and conclusions.

APLA-RAI-4

Please address the following questions associated with Attachment 1, "Seabrook Station PRA Peer Review Findings," of the May 31, 2016, LAR supplement.

a. The peer review finding for fact and observation (F&O) HR-G7-1 addresses the licensee's identification and treatment of dependency between multiple human actions. Please indicate if a specific floor value was defined (e.g., via post-processing) to ensure scenarios containing multiple human failure events/human error probabilities (HFEs/HEPs) did not drop below a minimum threshold. If any cutsets resulted in joint HEPs lower than 1E-6, provide a sensitivity evaluation of imposing such a minimum value and address whether this affects the conclusions drawn in the application.

b. The peer review finding for F&O 5-5 (IFSN-A9) addresses the potential for discrepancies between defined source values and associated spreadsheets. The reviewer provides an example where a turbine building flow rate of 15,000 gallons per minute is cited in a spreadsheet, whereas the source value was 56,000 gallons per minute. Please confirm the resolution of this discrepancy.

c. The peer review finding for F&O LE-E4-01 (SRs LE-E4 & E1) addresses the incorporation of state-of-knowledge uncertainty throughout the model. The licensee's resolution states that Level 1 and Level 2 sequences were reviewed to identify where the state-of-knowledge correlation might be important and noted that the ISLOCA evaluation explicitly accounts for the state-of-knowledge correlation. The licensee further states that based on its review, it is "judged" that other sequences would not benefit from application of state-of-knowledge correlation corrections. Please provide the basis for this judgment.

Mechanical and Civil Engineering Branch (EMCB)

EMCB RAI-1

In the licensee's letter dated March 31, 2016, Section 3.2.1 notes that Seabrook has no areas subject to American Society of Mechanical Engineers Boiler & Pressure Vessel Code (ASME Code), Section XI, Subsection IWE, augmented examinations. Section 3.2.1.1 notes that multiple indications on the containment liner were accepted by engineering evaluation during the last examination and require successive inspection per IWE-2420. ASME Code, Section XI, Subsection IWE, paragraph IWE-2420(b), states in part that when a component is acceptable based on engineering evaluation the area, "... shall be reexamined during the next inspection period ... in accordance with Table IWE 2500-1, Examination Category E-C [Containment Surfaces Requiring Augmented Examination]."

Please explain how areas can be identified for successive inspections per IWE-2420, yet the program can include a statement saying no areas are subject to augmented examinations.

EMCB RAI-2

Section 3.2.1.1 of the March 31, 2016, letter, summarizes recent inspections and corrective actions related to the containment liner to concrete floor moisture barrier. In the fall of 2015, degradation was identified of the moisture barrier and the liner near the moisture barrier. The degradation compromised the design function of the moisture barrier to seal the joint between the metal containment liner and the concrete floor slab. The licensee repaired the degraded moisture barrier and took ultrasonic testing (UT) measurements of the accessible areas of the metal liner. All UT measurements were above the nominal wall thickness, and the liner was recoated. These corrective actions adequately addressed identified degradation of the accessible portions of the liner and the moisture barrier. However, if the moisture barrier was degraded, and the accessible portion of the liner was degraded, it is likely liner degradation exists below the moisture barrier.

Title 10 of the Code of Federal Regulations (10 CFR) Section 50.55a(b)(2)(ix)(A) requires that the acceptability of inaccessible areas be evaluated when conditions exist in accessible areas that could indicate the presence of, or result in, degradation to such inaccessible areas. It is unclear to the NRC staff if this evaluation was completed, and how possible degradation of the inaccessible portions of the liner below the moisture barrier was addressed.

Please explain what was done to meet the requirements of 10 CFR 50.55a(b)(2)(ix)(A) and to demonstrate the acceptability of the inaccessible portions of the containment liner below the moisture barrier, or explain why no additional actions were necessary.

EMCB RAI-3

In Section 3.2.3 of the March 31, 2016, letter, the licensee summarized actions taken during the fall 2015 outage in response to NRC Information Notice 2014-07, "Degradation of Leak-Chase Channel Systems for Floor Welds of Metal Containment Shell and Concrete Containment Metallic Liner," related to inspections of leak-chase channel systems. The section includes a summary of the inspections conducted and the assumptions made in order to determine that the identified degradation was acceptable. It also includes an assumed corrosion rate of 0.0025 inches/year as well as an acceptable liner thickness of 1/8 inch. It is unclear to the NRC staff how the information provided led to the conclusion that the existing degradation was acceptable, and it is unclear how this issue will be addressed in the future. Please provide the additional information discussed below.

a. A tabular summary of the status of inspections conducted or planned for all 59 leak chase test connections. This should include how much of the connection was inspected in 2015 (i.e., outer cover, removed outer cover but inner plug stuck, video probe of riser to elbow, video probe of riser to leak chases, etc.) and summary of the results of the inspection. Connections that were not inspected should include a brief explanation of why they were not inspected, along with any plans to inspect the connections in the future.

b. A technical justification for the assumed corrosion rate of 0.0025 inches/year, including a discussion of the applicability of that assumed rate to the leak-chase channels.

c. The structural evaluation was only mentioned very briefly. Provide additional information on the evaluation, including the purpose of the original evaluation, the assumptions and limitations of the evaluation, and the applicability of the 1/8 inch limit (i.e., does the limit only apply to localized areas, or does it only apply to the portion of the liner in the floor).

d. An explanation of how leak chases will be inspected in the future under the ASME Code, Section XI, Subsection IWE program. This should include a discussion of whether or not the leak chases will be inspected during each period and if the leak chases that have been inspected to date will be subject to successive inspections per IWE 2420. Include a description of the acceptance criteria that will be used for leak chases.

EMCB RAI-4

Section 3.2.1.2 of the March 31, 2016, letter, provides a high level summary of the ASME Code, Section XI, Subsection IWL, inspection results for 2010 and notes that 84 suspect areas were identified that required engineering evaluation. The March 31, 2016, letter, also notes that walkdown assessments conducted under the Structures Monitoring Program identified four isolated locations of patterned cracking indicative of Alkali-Silica Reaction (ASR) on the containment. It is unclear if the ASR indications were identified within the 84 suspect areas noted during the 2010 IWL examinations. In order for the NRC staff to assess the proper and effective implementation of the ASME Code, Section XI, Subsection IWL, containment inspection program, please provide the following information:

a. Explain whether or not the ASME Code, Section XI, Subsection IWL, examination in 2010 noted the indications of ASR.

b. If the ASME Code, Section XI, Subsection IWL program did not identify the degradation, explain why not and what steps will be taken to ensure ASR indications on the containment will be identified and addressed in the future.

EMCB RAI-5

The licensee's letter dated May 31, 2016, provides a high level summary of ASR and the four ASR indications on the containment structure. The discussion provides an explanation of why containment leak-tightness should not be impacted by ASR. However, the discussion does not clearly address the ASR impact on structural integrity of the containment. Based on the ASR degradation, the containment is currently classified as operable, but degraded and non-conforming, which is currently an unresolved issue that may impact structural integrity.

Provide justification for extending the Type A test interval for the current interval (i.e., for the test due in 2018) without a positive physical verification of structural and leak-tight integrity in the current non-conforming condition.

EMCB RAI-6

Several sections in the March 31, 2016, letter, appear to have typographical errors or confusing terminology. Please address the issues identified below.

a. The second paragraph of the "Supplemental Inspection Requirement" in Section 3.2.2 mentions the performance of "containment structural integrity tests." Historically, a containment structural integrity test (SIT) is a pressure test of the containment at 1.15 design pressure. Please verify that this is not what was intended by the terminology in the LAR and clarify what the wording meant.

b. The final sentence in Section 3.2.4 states, "Additional detail on recent inspection is provided in Section 3.6.1." The LAR does not contain a Section 3.6.1. Please update the sentence with the correct reference.

Balance of Plant Branch (SBPB)

SBPB-RAI-1

Nuclear Energy Institute (NEI) Topical Report 94-01, Revision 0, "Industry Guideline for Implementing Performance Based Option of 10 CFR Part 50, Appendix J" (ADAMS Accession No. ML11327A025), reads, in part:

> 10.2.1.2, "Extended Test Intervals (Except Containment Airlocks)"
>
> The test intervals for Type B penetrations may be increased based upon completion of two consecutive periodic As-found Type B tests where results of each test are within a licensee's allowable administrative limits. Elapsed time between the first and last tests in a series of consecutive satisfactory tests used to determine performance shall be 24 months or the nominal test interval (e.g., refueling cycle) for the component prior to implementing Option B to Appendix J. An extended test interval for Type B tests may be increased to a specific value in a range of frequencies from greater than once per 24 months up to a maximum of once per 120 months. The specific test interval for Type B penetrations should be determined by a licensee in accordance with Section 11.0.

Letter dated March 31, 2016, Section 3.1.2 (first paragraph, Attachment 1, page 10 of 36), reads, in part:

> For Type B testing, 3 of 13 penetrations are currently on extended frequency. Two of the 13 penetrations are electrical penetrations. Since the electrical penetrations are train related, each penetration is tested every other refueling outage.

The NRC staff notes that this leaves eight of the Type B penetrations not on an extended test interval of, "from greater than once per 24 months up to a maximum of once per 120 months."

The NRC staff's review of the March 31, 2016, letter, "Table 2 - Type B Penetrations most recent two tests" (LAR Attachment 1, page 10 of 36), indicates that based on the "Limit (scfh [standard cubic feet per hour])" for each penetration, and the associated leakage values of the "Most Recent Test" and the "Previous Test," that all penetrations appear to be eligible, per the methodology in NEI 94-01, Revision 0, Section 10.2.1.2, to be on extended test intervals.

Based on the information contained in Table 2, the NRC staff requests that NextEra specifically describe: (a) which three penetrations are currently on an extended test interval of 120 months; (b) which penetrations are opened each refueling outage and, therefore, not eligible to be on an extended test interval; and (c) which penetrations are not included in parts (a) and (b) and explain why these residual penetrations are not on or eligible for an extended test interval (including any ongoing corrective actions).

SBPB-RAI-2

The March 31, 2016, letter, Section 3.1.2, "Type B and C Testing" (LAR Attachment 1, pages 8 and 9 of 36), details a history, dating back to March 1999 during OR06, of local leakage-rate test (LLRT) failures associated with inside containment isolation check valve, IA-V-531, for instrument air penetration X-68.

During OR06, the then existing valve was replaced with the soft seated check valve.

With the new soft seated valve, the following LLRT values were recorded during refueling outages:

OR06 (March 1999) – 0.279 scfh
OR07 – 0.542 scfh
OR08 – 0.589 scfh (IA-V-531 put on extended test frequency)
OR11 (fall 2006) – 1.492 scfh.
OR14 (spring 2011) – 4.592 scfh (Condition Reports 01673034/01682493 were Initiated to address the increasing trend)
OR15 (fall 2012) – 3.569 scfh (LLRT value after soft seat replacement via Work Order [WO] 40132878)
OR16 (spring 2014) – 10.661 scfh (value from LAR Table 3)
OR17 (fall 2015) – 19.661 scfh (value from LAR Table 3)

In Section 3.1.2, NextEra states that the LLRT failure(s), since installation of a soft seated check valve IA-V-531 in 1999, have been attributed to the following potential causes:

1. Dirt or grit from the carbon steel system in combination with close tolerances between the disc/disc guide and the bore; and/or
2. Advanced age of the soft seat.

The fourth paragraph of LAR Attachment 1, page 9 of 36, states:

> The age of the replacement seat may help explain why the AS LEFT test was non zero. A durometer test was not done, but the new soft seats felt slightly more pliable than the old soft seat.

The seventh paragraph of LAR Attachment 1, page 9 of 36, states:

> The local leak rate test was performed satisfactorily in OR16 and OR17, which re-establishes valve performance.

Based on the above, the NRC staff requests the following information:

a. Does the "AS LEFT test" value referred to in the fourth paragraph refer to the 3.569 scfh value listed above? The "age of the replacement seat," along with the large LLRT values associated with penetration X-68 since OR15, would suggest that the replacement seat installed under WO 40132878 had an advanced shelf life or was installed with an advanced shelf life. Given the carbons steel systems propensity for dirt and grit, and given the check valves' close tolerances, what justified installing an "aged" soft seat? An accurate interpretation of the fourth paragraph cited above is needed.

b. Since a soft seated check valve IA-V-531 was first installed in 1999, the penetration X 68 LLRT leakage rate has steadily increased and increased by a cumulative factor of more than 20 times the "as-left" leakage rate of OR06. What phenomena explains this steady increase in leakage rates? Are any corrective actions planned?

<u>SBPB-RAI-3</u>

NEI 94-01, Revision 0, states, in part:

> 10.2.3.2, "Extended Test Interval"
>
> Test intervals for Type C valves may be increased based upon completion of two consecutive periodic As-found Type C tests where the result of each test is within a licensee's allowable administrative limits. Elapsed time between the first and last tests in a series of consecutive passing tests used to determine performance shall be 24 months or the nominal test interval (e.g. refueling cycle) for the valve prior to implementing Option B to Appendix J. Intervals for Type C testing may be increased to a specific value in a range of frequencies from 24 months up to a maximum of 120 months. Test intervals for Type C valves should be determined by a licensee in accordance with Section 11.0.

Regulatory Guide 1.163 "Performance-Based Containment Leak-Test Program," September 1995 (ADAMS Accession No. ML003740058), Regulatory Position C.2, states:

> Section 11.3.2, "Programmatic Controls," of NEI 94-01 provides guidance for licensee selection of an extended interval greater than 60 months or 3 refueling cycles for a Type B or Type C tested component. Because of uncertainties (particularly unquantified leakage rates for test failures, repetitive/common mode failures, and aging effects) in historical Type C component performance data, and because of the indeterminate time period of three refueling cycles and insufficient precision of programmatic controls described in Section 11.3.2 to address these uncertainties, the guidance provided in Section 11.3.2 for selecting extended test intervals greater than 60 months for Type C

> tested components is not presently endorsed by the NRC staff. Further, the interval for Type C tests for main steam and feedwater isolation valves in BWRs, and containment purge and vent valves in PWRs and BWRs, should be limited to 30 months as specified in Section 3.3.4 of ANSI/ANS-56.8-1994, with consideration given to operating experience and safety significance.

Section 3.1.2 of LAR Attachment 1, page 10 of 36, second paragraph, states, in part:

> For Type C testing, 26 of 37 eligible penetrations are on extended frequency. This does not include the two penetrations that are required to be tested on a 30 month frequency per Regulatory Guide 1.163. Of the 11 penetrations not on extended frequency, four are train related and are tested every other outage. ... One penetration [i.e. X-68] is not on extended frequency due to the failure previously discussed. The remaining six penetrations are tested every outage.

The NRC staff notes that this leaves six of the Type C penetrations not on an extended test interval of greater than once per 24 months, up to a maximum of once per 60 months.

The NRC staff's review of "Table 3 - Type C Penetrations most recent two tests," (Attachment 1, pages 11 and 12 of 36), indicates that based on the "Limit (scfh)" for each penetration and the associated leakage values of the "Most Recent Test" and the "Previous Test" that all Penetrations appear to be eligible, per the methodology in NEI 94-01, Revision 0, Section 10.2.3.2, to be on extended test intervals.

Based on the information contained in Table 3, the NRC staff requests that NextEra specifically identify the six penetrations that appear to be eligible for extended test intervals but are not on an extended test interval. Provide a brief valve synopsis including description, design function, service life and any required corrective actions. Also provide an explanation of why these six penetrations are not on or have not qualified for an extended test interval.

SBPB-RAI-4

The NRC staff notes that the "Response for Seabrook" for Limitation Condition 3 in LAR Attachment 1 (page 25 of 36), contains an apparent error in the words, "Reference Section 3.2.1 through 3.2.9." The LAR only contains Sections 3.2.1 through 3.2.7. Please update the sentence with the correct reference.

October 3, 2016

Mr. Eric McCartney
Site Vice President
Seabrook Station
NextEra Energy
626 Lafayette Rd.
Seabrook, NH 03874

SUBJECT: SEABROOK STATION, UNIT NO. 1 – REQUEST FOR ADDITIONAL INFORMATION RE: REQUEST TO EXTEND CONTAINMENT LEAKAGE TEST FREQUENCY (CAC NO. MF7565)

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If you have questions, please contact me at 301-415-2048 or by e-mail at Justin.Poole@nrc.gov.

Sincerely,
/RA/

Justin C. Poole, Project Manager
Plant Licensing Branch I-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-443

Enclosure:
Request for Additional Information

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