NEXTera® ENERGY SEABROOK

October 27, 2016

10 CFR 50.90

SBK-L-16165

Docket No. 50-443

U. S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, DC 20555-0001

Seabrook Station

Response to Request for Additional Information Regarding License Amendment Request 16-01, "Request to Extend Containment Leakage Test Frequency"

References:

1. NextEra Energy Seabrook, LLC letter SBK-L-16029, "License Amendment Request 16-01, Request to Extend Containment Leakage Test Frequency," March 31, 2016 (ML16095A278)
2. NextEra Energy Seabrook, LLC letter SBK-L-16082, "Supplement to License Amendment Request 16-01, Request to Extend containment Leakage Test Frequency," May 31, 2016 (ML16159A194)
3. NRC letter "Seabrook Station, Unit No. 1 – Request for Additional Information Related to Request to Extend Containment Leakage Test Frequency," October 3, 2016 (ML16230A106)

In Reference 1, NextEra Energy Seabrook, LLC (NextEra) submitted a license amendment request (LAR) 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".

In Reference 3, the NRC staff determined that additional information is necessary to support the staff's review of the proposed change. The enclosures to this letter provide the requested information.

This response to the request for information does not alter the conclusion in Reference 1 that the change does not involve a significant hazards consideration pursuant to 10 CFR 50.92, and there are no significant environmental impacts associated with this change.

No new or revised commitments are included in this letter.

A001
NRR

Should you have any questions regarding this letter, please contact Mr. Kenneth Browne, Licensing Manager, at (603) 773-7932.

I declare under penalty of perjury that the foregoing is true and correct.

Executed on October 27, 2016.

Sincerely,

Eric McCartney
Site Vice President
NextEra Energy Seabrook, LLC

Enclosures:

Enclosure 1 Response to Request for Additional Information Regarding License Amendment Request 16-01, "Request to Extend Containment Leakage Test Frequency"

Enclosure 2 Changes to Attachment 4, "Risk Impact Assessment" from NextEra Energy Seabrook, LLC letter SBK-L-16029, "License Amendment Request 16-01, Request to Extend Containment Leakage Test Frequency, March 31, 2016 (ML16095A278 and ML16159A194)

cc: NRC Region I Administrator
NRC Project Manager
NRC Senior Resident Inspector

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Enclosure 1 to SBK-L-16165

Response to Request for Additional Information Regarding License Amendment Request 16-01, “Request to Extend Containment Leakage Test Frequency”

Background

By letter dated March 31, 2016 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML 16095A278), as supplemented by letter dated May 31, 2016 (ADAMS Accession No. ML 16159A 194), 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 RateTesting 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.

The NRC staff has determined that additional information provided below is necessary to complete the review.

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. ML 15209A870). 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.

NextEra Response to APLA-RAI-1

The Seabrook PRA results presented in NUREG1437, Supplemental 46 are based on the Seabrook Station PRA model issued in 2011. The PRA model was updated in 2014. The updated at-power total CDF (from all internal and external events) decreased by approximately 3E-07/yr, from 1.23E-05/yr to 1.20E-05/yr mean value. This represents a CDF decrease of ~2.5%.

Table 1 provides a comparison of the 2011 quantitative Level 1 results to the 2014 (ILRT) results for internal events, internal flooding and external flooding. To clarify, it is noted that the

CDF value of 7.8E-06 per year summarized in NUREG-1437, Supplement 46 also included the CDF contribution from severe weather events.

Table 1 - Comparison of 2011 to 2014 Level 1 CDF Results

Hazard	CDF/yr 2014	CDF/yr 2011	Delta CDF/yr (2014-2011)	Change Fraction (from 2011)
Internal Events	3.76E-06	4.50E-06	-7.40E-07	-0.16
External Flood	2.86E-08	2.40E-08	4.60E-09	0.19
Internal Flood	2.86E-06	2.61E-06	2.50E-07	0.10
severe weather	6.36E-07	6.82E-07	-4.60E-08	-0.07
Total	**7.28E-06**	**7.82E-06**	-5.31E-07	**-0.07**

The difference in CDF between the 2011 and 2014 Level 1 models for internal events, internal flooding, external flooding and severe weather reflects a CDF reduction of approximate 7%. This change in CDF is attributed to the modeling changes made in the 2014 PRA update.

Table 2 provides a comparison of the 2011 quantitative Level 2 results to the LAR Table 4-2 (2014) Level 2 results for all at-power events and hazards. The updated at-power LERF increased by 6.3E-08/yr, from 9.2E-08/yr to 1.55E-07/yr mean value. Although LERF remains relatively low, the updated value represents an increase of ~68%.

Table 2 - Comparison of 2011 to LAR Table 4-2 Level 2 Release Results

Release Category	Table 4-2 2014 ReleaseFreq/yr[1]	2011 Release Freq/yr	Delta Release (2014-2011)	Change Fraction (from 2011)
INTACT1	7.87E-06	7.07E-06	8.00E-07	0.11
INTACT2	7.73E-08	6.90E-08	8.30E-09	0.12
SE1	5.56E-07	5.08E-07	4.80E-08	0.09
SE2	2.82E-08	2.79E-08	3.00E-10	0.01
SE3	2.30E-07	9.97E-07	-7.67E-07	-0.77
LL3	2.71E-07	1.75E-07	9.60E-08	0.55
LL4	1.03E-06	5.79E-08	9.72E-07	16.79
LL5	9.99E-07	3.10E-06	-2.10E-06	-0.68
SELL	7.50E-07	9.84E-08	6.52E-07	6.62
LE1	4.36E-08	5.19E-08	-8.30E-09	-0.16
LE2	1.81E-08	1.81E-08	0.00E+00	0.00

Table 2 - Comparison of 2011 to LAR Table 4-2 Level 2 Release Results				
Release Category	**Table 4-2 2014 ReleaseFreq/yr(1)**	**2011 Release Freq/yr**	**Delta Release** (2014-2011)	**Change Fraction** (from 2011)
LE3	8.59E-10	8.61E-10	-2.00E-12	0.00
LE4	9.20E-08	2.11E-08	7.09E-08	3.36
Total	**1.20E-05**	**1.22E-05**	-2.29E-07	**-0.02**

Note (1): 2014 releases taken from Table 4-2 of Attachment 4 to Seabrook ILRT LAR

Table 3 provides a comparison of the 2011 quantitative Level 2 results to LAR Table 4-3 (2014) Level 2 Release Category Summary. This table compares the summary of releases binned for the ILRT evaluation. The net change in each bin total is consistent with the changes described above.

Table 3 - Comparison of 2011 to LAR Table 4-3 Release Summary				
Release Category	**Table 4-3 2014 Freq/yr (2)**	**2011 Freq/yr**	**Delta Release** (2014 - 2011)	**Change Fraction** (from 2011)
INTACT	7.95E-06	7.14E-06	8.08E-07	0.11
SERF	8.15E-07	1.53E-06	-7.18E-07	-0.47
LATE	3.05E-06	3.43E-06	-3.83E-07	-0.11
LERF	1.55E-07	9.20E-08	6.25E-08	0.68
Total	1.20E-05	1.22E-05	-2.31E-07	-0.02

Note (2): 2014 release summary taken from Table 4-3 of Attachment 4 to Seabrook ILRT LAR

The Level 1 and Level 2 PRA results cited in Tables 4.1, 4.2 and 4.3 of Attachment 4 to Seabrook's ILRT LAR represent the most current plant-specific, best estimate information for Seabrook Station. Therefore, the information in the LAR tables is the most relevant to the ILRT application, meets the order of preference suggested in the EPRI guidance, and no changes are judged necessary to further improve upon the plant specific nature of the inputs.

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.

NextEra Response to APLA-RAI-2

The discrepancy identified in this RAI is an error in the calculation of the "frequency-weighted" population dose for Release Classes 7 and 8, which resulted in the population doses for these release classes to be under predicted. For the Class 7 weighted population dose, the assessment should not have included the frequency and dose of release category SE-3. In addition, for the Class 8 weighted population dose, the assessment inadvertently omitted the frequency and dose of release categories LE-1 and LE-2. Although correction of these errors caused an increase in the Class 7 and Class 8 population doses, the increase has a negligible effect on the overall change in population dose (delta total dose rate) for the ILRT interval at 15 years and there is no change to the ILRT risk conclusions. The affected pages of LAR Attachment 4 have been marked-up and attached under Enclosure 2 to this RAI response letter. The pages in LAR Attachment 4 affected by this RAI include: pages 25, 27, 28, 30, 31, 33, 34, 36, 39, 43, 44 and 46.

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.

NextEra Response to APLA-RAI-3

The following provides additional explanation of the fire events contribution to LERF based on the current 2014 model of record for Seabrook Station. Fire events LERF is captured in release categories LE1 and LE3.

LE1 has a very small fire CDF contribution of approximately 8E-14, which is caused by a thermally-induced SGTR subsequent to core damage. Otherwise, LE1 releases capture steam generator tube rupture (SGTR) events, which are not initiated by internal fire events.

LE3 releases capture the remainder of the internal fire events LERF and have a total release frequency of ~1.3E-10/yr. The LE-3 bin is used to capture release sequences that involve a large containment penetration failure-to-isolate. The containment penetration failure is a random failure of the Containment Operating Purge System (COP) valves. The COP penetrations consist of a ventilation supply and exhaust path, with each path equipped with two, fail closed, 8-inch, pneumatically-operated valves (one inboard and one outboard). The outboard isolation valves are powered from a divisional power source separate from the power source to the inboard valves. The COP valves fail-closed on loss of air/power and historically are only opened during power operation approximately 10 percent of the time. Based on the above, the COP penetration isolation design is very reliable.

The assessment of the fire-induced impact on containment isolation concluded that the impact of fires on the containment performance is to fail a single train of isolation. For isolation failure of one or more valves of a single train, either the redundant isolation valve would remain available or the ability to remove power from failed closed valves is available to provide isolation. Thus, the overall conclusion reached in the review of fire induced impact on containment isolation is that the frequency of fires that could cause sufficient damage to fail containment isolation is low enough, compared to hardware failures, to not contribute significantly to loss of containment function. This coupled with the low percentage of time that the large CI valves are open during power further supports that a fire-induced large containment isolation failure is of very low probability.

Release Category LE2 captures the large-early release frequency only associated with ISLOCA-type sequences. These sequences are initiated by passive and catastrophic mechanical failure of redundant pressure isolation valves consisting of redundant normally closed check valves and redundant train, normally closed motor-operated valves (MOVs).

During normal plant power operation, the motive and control power for the normally closed MOVs is de-energized by locked-open power breakers. Given the de-energized state of these MOVs and the separate train power supplies, it is considered not credible for a fire event to initiate an ISLOCA-type scenario. Therefore, LE2 releases are not initiated by internal fire.

Release Category LE4 captures release sequences involving extreme seismic events where offsite emergency protective actions (evacuation) could be significantly delayed. The LE4 releases are not initiated by internal fire.

The majority of the fire-initiated core damage release sequences are large-late (LL) sequences resulting from long term containment over-pressure failure and basemat melt-through failure. This is consistent with the overall Level 2 results for all other initiating events that are not SGTR, ISLOCA or extreme seismic events. Thus, given the above, fire-initiated core damage events have roughly the same impact with regard to containment response as other non-SGTR/ISLOCA internal events.

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.

NextEra Response to APLA-RAI-4

Part (a):

The Seabrook RISKMAN PRA model/approach does not include "HEP post-processing". Instead, dependence among human actions is accounted for within the baseline PRA model as determined by the HEP dependency analysis. This approach uses predetermined HEP values and associated assignment rules to capture the level of HEP dependence (complete, partial or zero dependence) among the multiple HEPs in a sequence. Implementation of a specific minimum floor value of 1E-06 for joint HEPs in a sequence is not included in the baseline PRA model.

As requested in the RAI, a sensitivity evaluation was performed to determine the potential impact on the ILRT conclusions when limiting the sequence total joint HEP value to no less than 1E-06. In the sensitivity evaluation no attempt was made to justify a lower minimum HEP floor value where the state of HEP independence may have been justified by the HEP dependency analysis, thus providing a conservative application of the assumed HEP floor evaluation. In addition, the sensitivity evaluation conservatively applied the floor value for all sequences containing multiple HEPs down to a sequence truncation level of E-14. The following insights were noted:

1. The top level sequences (many thousands of sequences) contributing to CDF/LERF contain very few instances of joint HEPs. These sequences are negligibly impacted by the assumed floor value.
2. Lower level sequences (many hundreds of thousands) contain more instances of joint HEPs and these sequences are affected more by the application of the floor value than the higher level sequences. 3. At a sequence truncation of E-14, CDF is approximately 2.50E-05/yr, an increase of 109% above the baseline value of 1.20E-05/yr. LERF is approximately 1.65E-07/yr, an increase of 7% above the baseline value of 1.55E-07/yr. In addition, the ILRT Class 2 release frequency increased to 2.43E-07/yr (from 2.31E-07/yr) and the Class 8 release frequency increased to 1.28E-06/yr (from 6.46E-07/yr).

This sensitivity evaluation shows that the 15-year ILRT test interval Class 3b LERF is 2.72E-07/yr resulting in a delta-LERF of 2.17E-07/yr. This represents an increase in the delta-LERF of approximately 1.15E-07/yr above the original ILRT delta-LERF of 1.02E-07/yr. The total LERF is approximately 1.65E-07/yr + 2.72E-07/yr = 4.37E-07/yr. In addition, the delta-total dose rate is 7.93E-02 person-rem/yr and conditional containment failure probability is 0.87%.

Conclusion: The sensitivity evaluation suggests that the ILRT delta-LERF is sensitive to the assumption of the HEP floor value of 1E-06. The sensitivity ILRT delta-LERF of 2.17E-07/yr is just above the 1E-07/yr acceptance criterion and the total LERF of 4.37E-07/yr is well below the 1E-05/yr acceptance criterion. These changes in LERF would be defined as small per Region II of Regulatory Guide 1.174. The additional ILRT risk metrics of delta-total dose rate at 7.93E-02 person-rem/yr and conditional containment failure probability at 0.87% also remain well below the ILRT acceptance criteria of less than 1 person-rem/yr and less than 1.5% respectively.

Part (b):

Internal flooding F&O F&O 5-5 (IFSN-A9) identified a potential documentation issue and there were no discrepencies in the assumed flooding flow rates used to develop the associated scenarios. The proper internal flooding flow rates estimated for the various flood sources in the plant, including the Turbine Building, are used in the development of the associated flooding scenarios.

Part (c)

The original judgment that other sequences would not benefit from application of the state-of-knowledge correlation corrections was based on a review of sequences and associated group contributions. This was further supported by a check of Monte Carlo (MC) simulation uncertainty results for selected major top event frontline mitigation systems. The check did not identify significant differences between the split fraction MC-generated mean values and the point estimate mean values. This suggested that the key contributors to the selected top events are not particularly sensitive to, or do not involve, multiple occurrences of the same variable. Therefore, the SOKC was judged to be adequately addressed.

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.

NextEra Response to EMCB RAI-1

At the time of the initial submittal dated March 31, 2016, the IWE Program was being revised to include areas subject to augmented inspections per IWE-1241(c), and these areas are scheduled to be re-examined in accordance with Table IWE-2500-1. The statement in the submittal was based on the previous revision of the program reflecting the status at the end of the first period. The augmented inspections are now included in the program.

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.

NextEra Response to EMCB RAI-2

The analysis for the containment liner identifies that the cylinder portion of the liner plate has a nominal wall thickness of 3/8" (0.375"), a general corrosion allowance of 3/32" (0.094") and a local corrosion allowance of 3/16" (0.1875") not to exceed a liner surface area of 10 sq. ft.

During OR16 in 2014, fifty-one locations of the internal containment liner at the minus 26 ft (-26') location adjacent to the concrete-to-metal interface moisture barrier were examined by ultrasonic examination. The thickness readings ranged from 0.368" to 0.408" in thickness, which is within the 0.010" permissible variation under the specified thickness in SA-20.

During OR17 in 2015 an additional thirty-six areas of the containment liner at the concrete-to-metal interface moisture barrier were selected for ultrasonic examination due to surface corrosion of the liner during the moisture barrier exam. The lowest thickness measurements ranged from 0.378" to 0.408". The results of the thickness readings in the accessible areas in 2014 and 2015 indicate there is no degradation below the nominal specified thickness with respect to permissible tolerance.

There was no measurable corrosion degradation of the liner in the accessible areas below the 10% criteria in IWE-3511.3. However,since there was degradation of the moisture barrier in the concrete-to-metal interface, and moisture could have migrated past the degraded moisture barrier, a corrosion evaluation of the inaccessible is warranted. As stated above, the general corrosion allowance for the cylindrical corrosion liner is 3/32" (0.094"). By comparison, the corrosion allowance for the liner plate is 1/8" (0.125"). The conditions for the inaccessible area below the moisture barrier are similar to the base mat since the liner is adjacent to the concrete and oxygen is limited. Therefore, it is reasonable to assume the corrosion rate for the inaccessible area below the moisture barrier would be the same as the corrosion rate for the base mat. Since the corrosion allowance for the base mat is less than the cylindrical liner plate, the corrosion analysis is bounded by the base mat analysis and further justified in the response to EMCB RAI-3 below.

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.

NextEra Response to EMCB RAI-3

Part (a):

The following table provides a summary of the extent of the leak chase inspections. Note that there are only 57 leak chase access locations at Seabrook. Those locations that were not inspected will be inspected during the next outage in the spring of 2017.

Penetration Number	Type	Outer cover/plug inspected	Inner plug inspected	Video Scope	Results or Reason for not inspecting
HL 1-1	Floor Plug	X			Note 4 Could not remove outer cover
HL 1-2	Floor plug	X			Note 4 Could not remove inner plug
HL 1-3	Floor Plug	X			Note 4

Penetration Number	Type	Outer cover/plug inspected	Inner plug inspected	Video Scope	Results or Reason for not inspecting
					Could not remove outer cover
HL 1-4	Floor Plug	X	X	Leak chase	Note 4
HL 1-5	Floor plug	X			Note 4 Could not remove inner plug
HL 1-6	Floor plug	X	X	Leak chase	Note 4
HL 1-7	Floor Plug	X			Note 4 Could not remove outer cover
HL 1-8	Floor plug	X			Note 4 Could not remove inner plug
HL 1-9	Floor plug	X	X	Riser only	Notes 1,4
HL 1-10	Floor plug	X	X	Riser only	Notes 1,4
HL 1-21	Floor Plug	X			Note 4 Could not remove outer cover
HL 1-22	Floor plug	X			Note 4 Could not remove inner plug
HL 1-23	Floor Plug	X			Note 4 Could not remove outer cover
HL 1-24	Floor plug	X	X	Leak chase	Note 4
HL 1-25	Floor plug	X	X	Leak chase	Note 4
HL 1-27	Floor plug	X	X	Riser only	Notes 1, 4
HL 2-11	Floor plug	X			Note 4 Could not remove inner plug
HL 2-12	Floor plug	X	X	Leak chase	Note 4
HL 2-13	Floor plug	X			Note 4 Could not remove inner plug
HL 2-14	Floor Plug	X			Note 4 Could not remove outer cover
HL 2-15	Floor plug	X			Note 4 Could not remove inner plug
HL 2-16	Floor plug	X	X	Leak chase	Note 4
HL 2-17	Floor plug	X	X	Riser only	Notes 1 and 4
HL 2-18	Floor plug	X	X	Riser only	Notes 1 and 4
HL 2-19	Floor Plug				Note 4 Outer plug inaccessible
HL 2-20	Floor plug	X			Note 4 Could not remove inner plug
HL 3-1	Access channel	X	N/A	N/A	Note 2
HL 3-2	Access channel	X	N/A	N/A	Note 2
HL 3-3	Access channel	X	N/A	N/A	Note 2

Penetration Number	Type	Outer cover/plug inspected	Inner plug inspected	Video Scope	Results or Reason for not inspecting
HL 3-4	Access channel	X	N/A	To liner	Note 3
HL 3-5	Access channel	X	N/A	N/A	Note 2
HL 3-6	Access channel	X	N/A	N/A	Note 2
HL 3-7	Access channel	X	N/A	N/A	Note 2
HL 3-8	Access channel	X	N/A	N/A	Note 2
HL 3-9	Access channel	X	N/A	N/A	Note 2
HL 3-10	Access channel	X	N/A	N/A	Note 2
HL 3-11	Access channel	X	N/A	N/A	Note 2
HL 4-1	Floor Plug	X			Note 4 Could not remove outer cover
HL 4-2	Floor Plug	X			Note 4 Could not remove outer cover
HL 4-3	Floor Plug	X			Note 4 Could not remove outer cover
HL 4-4	Floor Plug	X			Note 4 Could not remove outer cover
HL 4-5	Floor Plug	X			Note 4 Could not remove outer cover
HL 4-6	Floor Plug	X			Note 4 Outer cover inaccessible
HL 4-7	Floor plug	X	X	Riser only	Notes 1,4
HL 4-8	Floor plug	X	X	Riser only	Notes 1,4
HL 4-9	Floor plug	X	X	Riser only	Notes 1,4
HL 5-1	Floor plug	X	X	Riser only	Notes 1,4
HL 5-2	Floor plug	X	X	Riser only	Notes 1, 4
HL 5-3	Floor plug	X	X	Riser only	Notes 1, 4
HL 5-7	Floor plug	X	N/A	N/A	No degradation found under cover
HL 5-8	Floor Plug	X			Note 4 Could not remove outer cover
HL 5-9	Floor plug	X	X	Riser only	Notes 1,4
HL 5-11	Floor plug	X	X	Riser only	Notes 1,4
HL 6-2	Access channel	X	N/A	N/A	Note 2
HL 6-3	Floor Plug	X			Note 4 Could not remove outer cover
HL 6-4	Access channel	X	N/A	N/A	Note 2

Notes:
1. Video inspection could not reach leak chase due to presence of elbows
2. Access channels extend above the floor slab and are not susceptible to water intrusion. These locations were acceptable. No degradation noted.
3. Some flaking paint found at this access channel Removed plug and performed video inspection of liner. No degradation identified.
4. Evaluation of identified degradation (moisture and/or corrosion) is provided in response to RAI-3 Part b below.

Part (b):

Chemistry samples were taken at the leak chase locations where water was present. The sample results had consistent elevated sample pH values between 12.5 to 13.1and iron concentration values between 2 to 5 ppm, indicating the water was stagnant and in intimate contact with concrete for a long time period, likely several years. Based on the chemistry results, it was determined that the steel liner and chase surfaces are protected with a tight adherent iron hydroxide Fe(OH)2 corrosion layer (or film) that is in equilibrium with the water above the metal surface. This determination was made based on information available in DOE Chemistry Fundamentals Handbook DOE-HDBK-1015/2-93 and Sandia Report SAND2010-8718. All leak chase samples have chloride and sulfate concentrations that are less than several hundred ppm. These concentration levels indicate that the iron hydroxide passive layer has not been disrupted. Accordingly, the base metal has not been adversely impacted and is in a stable condition. Based on the sample pH values alone, the estimated corrosion rate of 0.0025 inches per year, or 0.075 inches maximum total corrosion for a 30 year exposure is obtained. This corrosion rate is not indicative of the stagnant water conditions in the leak chases. The corrosion rates that would be experienced in this area should be significantly lower than figures identified in the literature. The conditions required for corrosion to take place are a bimetallic substrate, an electrolyte, and oxygen. If anyone of these elements is below threshold values, then the corrosion would be less than what is shown in the literature. The lack of movement of the water contained within the channels would prevent erosion of the passive corrosion layer. Corrosion rates lower than 0.0025 inches per year are supported by boroscope inspections of the accessible areas. The boroscope inspections indentified minor scale and a very clean metal surface with a minor amount of corrosion present. However, as a conservative measure, the estimated worst case corrosion rate of 0.0025 inches per year was applied to assess total corrosion in the leak chase channels.

Part (c):

The purpose for the evaluation was to satisfy the requirements of ASME Section XI Subsection IWE paragraph IWE-3122.3 for acceptance of examination results by engineering evaluation. The evaluation documents acceptance of the identified conditions for continued service without repair or replacement. The evaluation concludes that the identified degradation in the containment liner meets the minimum wall thickness and has no unacceptable effect on the structural integrity of the containment. A minimum wall thickness of the liner is required to ensure it will remain leak tight post-accident. This evaluation also bounds the inaccessible areas in the leak chase channel for continued operation.

The liner nominal wall thickness is 3/8" or 0.375" in the containment cylindrical portion and the base mat or leak chase channel nominal wall thickness is 1/4" or 0.250".

The minimum wall thickness for the liner under the floor or in the leak chase channels is 1/8" or 0.125". In the 2015 engineering evaluation, the leak chase channel systems are independent areas of the rest of the containment liner. The liner in the leak chase channels is categorized as E-C in the IWE program. The total surface area inside the leak chase channels is less than 2000 square feet. The conclusion from the evaluation is that a minimum liner wall thickness of 1/8" is acceptable for the areas inside the leak chase channels provided the accumulated surface area does not exceed 2000 square feet. The estimated corrosion rate in the leak chase channels is based on the observed chemistry from the water samples, which is 0.0025 inches per year (Reference RAI-3 Part b). The visual inspections of sample liner locations in the leak chase channels did not identify a significant loss of material, which supports the maximum wear rate applied. The projected liner wall thickness in the leak chase channels meets the allowable minimum thickness for at least 50 years of operation.

The metallic liner of containment is credited as being a leaktight membrane. UFSAR 3.8.1.1 states "A welded steel liner plate, anchored to the inside face of the containment, serves as a leak tight membrane."

The minimum wall thickness in the containment cylindrical portion, except the leak chase channels, is 3/16", and it is only acceptable in a contiguous area not to exceed 10 square feet. Applying the maximum wear rate (0.0025 inches per year) to the inaccessible and accessible portions of the cylindrical liner area meets the allowable minimum thickness for at least 75 years of operation.

There are no unacceptable conditions on the structural integrity or on the leak tight membrane of the containment structure of the examined and unexamined areas. The visual inspection of the sample liner locations did not identify a significant loss of material.

Part (d):

The leak chase channel plugs are considered moisture barriers (Table IWE-2500-1, Category E-A, Item E1.30) and will be inspected each inspection period. The leak chases and associated risers are considered surface areas requiring Augmented examination (IWE-1241 (a) and will be examined in accordance with Table IWE-2500-1, Category E-C, Item No. E4.11. For leak chases with risers that allow access to the leak chase channel, remote visual examination will be performed to the extent possible. The leak chase channel plugs and leak chase channels that have been inspected to date are subject to successive examinations per IWE-2420, Table IWE-2500-1, Category E-C, and are scheduled to be inspected in the next period.

Moisture barriers with wear, damage, erosion, tear, surface cracks, or other defects that permit intrusion of moisture to inaccessible areas of the pressure retaining surfaces of the metal containment shell or liner shall be corrected. The affected inaccessible areas will be inspected

using remote methods, when possible, and evaluated to address potential degradation against design requirements.

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.

NextEra Response to EMCB RAI-4

a. Indications of ASR noted on containment were not identified during the ASME Code, Section XI, Subsection IWL, examination completed in 2010. The IWL inspection procedure in place at that time did not include inspection requirements for containment concrete surfaces to identify the presence of ASR.

b. The initial discovery of ASR at Seabrook was in the Control Building in August 2010. The ASME Code, Section XI, subsection IWL examination completed in September of 2010 occurred during the discovery phase of ASR when the extent and scope of structures affected by ASR was not known. As a result, the 2010 ASME IWL examination of containment did not include requirements for identifying the presence of ASR. The presence of ASR in containment was documented in September 2012 as part of the ASR walkdowns completed under the Structural Monitoring Program. The Seabrook ASME IWL examination procedure was revised in April 2013, and included the addition of inspection requirements for containment concrete surfaces to identify the presence of ASR. The 2016 Section XI, subsection IWL examination has been completed and utilized the revised ASME IWL examination procedure that includes a requirement to inspect for the presence of ASR on containment concrete surfaces. Any indications of ASR on containment identified during ASME IWL examinations are evaluated in accordance with Seabrook's Structural Monitoring Program and ASME Section XI, subsection IWL, paragraph IWL-3300.

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 nonconforming, 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.

NextEra Response to EMCB RAI-5

The containment building is a reinforced concrete structure that is designed in accordance with the requirements of Section III of the American Society of Mechanical Engineers (ASME) Boiler & Pressure Vessel Code (1975 Edition). The Seabrook containment was evaluated for structural and leak-tight integrity and determined to be operable, but degraded and nonconforming as it is not in compliance with the ASME code because the ASME code does not include provisions for the analysis of containments affected by ASR.

The containment structure is comprised of two major structural elements, the biological shield portion (concrete portion) and the gas barrier (steel liner). The biological shield is the structural element of the containment and is constructed of reinforced concrete. The gas barrier is constructed of carbon steel plate referred to as the liner which acts as a leak-tight barrier. The containment integrated leakage rate test (ILRT - Type A test) is performed to verify and demonstrate leak tightness of the steel liner. The concrete portion of containment was evaluated for the effects of ASR and determined to maintain structural integrity and have adequate margin to meet all of its design basis functions. During the 16th refueling outage (OR16) the containment liner was ultrasonically examined to determine if wall loss has occurred due to corrosion on the opposite surface (concrete side). The examination was performed in response to a license renewal commitment. Examination areas included areas that were local to the 4 locations identified with ASR on the concrete surface of containment identified in 2012. The examination areas did not exhibit any signs of corrosion or metal loss on the opposite (concrete) side of the steel liner.

Therefore, verification of structural and leak-tight integrity of containment for the concrete areas affected by ASR identified in 2012 is provided based on the structural evaluation completed for the concrete and the results of the ultrasonic examination of the liner.

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.

NextEra Response to EMCB RAI-6

a. The "containment structural integrity tests" should be changed to the "containment inspections". These inspections are directed by the Containment and Containment Enclosure Surface Inspection Procedure for Appendix J and the Primary Containment Section XI Inservice Inspection Procedure for the IWE/IWL program.

b. Section 3.6.1 should be 3.2.6

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).

NextEra Response to SBPB-RAI-1

(a) Which three penetrations are currently on an extended test interval of 120 months:

All Type B penetrations are considered eligible for extended testing, however some Type B tests either have scheduled maintenance tasks that requires frequent testing, or SBK needs to make changes to the LLRTs PMs in order to change the test interval.

- H2 Analyzer Train A – last tested in June 2008.
- H2 Analyzer Train B- last tested in October 2014. Testing tied to maintenance activities.
- Large diameter Equipment Hatch (removable) Flange Seals – tested whenever the hatch is removed for equipment access – tested the last three outages, however before that there was one outage that it was not removed and not tested.

(b) Which penetrations are opened each refueling outage and therefore not eligible to be on an extended test interval:

The penetrations that are opened each refueling outage are:

- HVAC 1 Flange
- HVAC 2 Flange
- Fuel Transfer Tube Flange
- Equipment Hatch Flange Seals – dependent of moving large equipment in and out of containment.
- Spare (E-58) – Opened during outages for services.
- Spare (E-59) – Opened during outages for services.

(c) Which penetrations are not included in parts (a) and (b) but are eligible for extended test intervals and explain why these residual penetrations are not on an extended test interval (including any ongoing corrective actions):

The penetrations not included in parts (a) and (b) are:

- Fuel Transfer Tube (FTT) Metal Bellows – tested in April 2014 and Oct 2009.
- Electrical Penetrations 1 Train A (# 1-33) – tested every other outage as group in one PM task.
- Electrical Penetrations 2 Train B (#33 – 56) – tested every other outage as group in one PM task.

These penetrations are on an extended frequency but not on an extended frequency of 120 months. There are no ongoing corrective actions in regards to these penetrations.

Electrical Penetrations 1 and 2 are Train Related with a mixture of high voltage, medium voltage and instrumentation penetrations. They are tested every other refueling outage. The site is

evaluating how this testing can be further divided to allow the scheduling programmatic controls to be applied and ensure testing at approximately even – distributed intervals.

SBPB-RAI-2

The March 31, 2016, letter, Section 3.1.2, "Type Band 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 OR 16 and OR 17, which reestablishes valve performance.

Based on the above, the NRG 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?

NextEra Response to SBPB-RAI-2

a(1) Does the "AS LEFT test" value referred to in the fourth paragraph refer to the 3.569 scfh value listed above?

Yes. The "AS LEFT test" value referred to in the fourth paragraph is 3.569 scfh. This test was performed in the fall of 2012. The "AS FOUND test" was unable to hold pressure.

a(2) Given the carbon steel systems propensity for dirt and grit and given the check valves close tolerances, what justified installing an "aged" soft seat?

The successful as-left test of 3.569 scfh justified that the aged soft seat would perform the intended function of containment isolation within the sites Administrative Limit of 147.8 scfh. There were no restrictions provided by the manufacturer on the shelf life or hardness requirements for a new seat. The qualitative determination that the new seat was more pliable than the removed seat was used to justify the installation of the new seat. This decision was confirmed by the as-left test result. The penetration is currently tested every outage.

a(3) An accurate interpretation of the fourth paragraph cited above is needed.

"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."

As the soft seat ages the pliability decreases and the seat losses some ability to conform to the mating seat. The replacement seat was found to be slightly more pliable than the seat that was removed resulting in a lower as-left leakage. However, the replacement seat was not as pliable as a recently manufactured seat and was therefore, more susceptible to leakage. The valve will be tested in the Spring 2017 outage and repaired as necessary.

(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?

The most likely cause of the increasing leakage rate is normal wear on the soft seats. The corrective action was to write a PM for the disassembly of the check valve for the inspection of internals and for soft seat replacement every fourth refueling outage. This penetration is leak rate tested every refueling outage. The valve will be leakage rate tested and repaired as necessary. A work request has been written as a contingency work order if a repair is required in OR18.

SBPB-RAI-3

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.

NextEra Response to SBPB-RAI-3

1. Specifically identify the six penetrations that appear to be eligible for extended test intervals but are not on an extended test interval:

   The six penetrations not on extended frequency are:

   - X-16, valves COP-V3 - and COP-V4
   - X-18, valves COP-V2 and COP-V1
   - X-35C, valves RC-FV2874 and RC-FV2894/RC-FV2832 and RC-V314
   - X-35D, valves RC-FV2876 and RC-FV2896/RC-FV2833 and RC-V337
   - X-38A, valves FP-V588/FP-V592
   - X-38B, valves CGC-V46/CGC-V43 and CGC-V44 and CGC-V45

2. Provide a brief valve synopsis including description, design function, service life and any required corrective actions:

Please see the following table:

Valve Synopsis

Penetration	Valve	Description	Design Function	Service Life	Corrective Actions
X-16	COP-V3	Containment online purge exhaust isolation valve IRC-CIV 8" Posi Seal butterfly valve with Matryx pneumatic actuator	This valve may be open during power operation to provide filtered air for purging the containment (manually controlled by operator to adjust containment pressure to 0.50 +/- 0.15 psig) and receives a Containment Ventilation Isolation Signal (CVIS) to close.	OR13 – 2.262 scfh OR14 – 1.759 scfh OR15 – 1.510 scfh OR16 – 1.603 scfh OR17 – 1.638 scfh Acceptance Criteria: 7.4 scfh	There are no open corrective actions.
	COP-V4	Containment online purge exhaust isolation valve ORC-CIV 8" Posi Seal butterfly valve with Matryx pneumatic actuator	This valve may be open during power operation to provide filtered air for purging the containment (manually controlled by operator to adjust containment pressure to 0.50 +/- 0.15 psig) and receives a Containment Ventilation Isolation Signal (CVIS) to close.	OR13 – 2.262 scfh OR14 – 1.759 scfh OR15 – 1.510 scfh OR16 – 1.603 scfh OR17 – 1.638 scfh Acceptance Criteria: 7.4 scfh	There are no open corrective actions.
X-18	COP-V2	Containment online purge supply isolation valve IRC-CIV 8" Posi Seal butterfly valve with Matryx pneumatic actuator	This valve may be open during power operation to provide filtered air for purging the containment (manually controlled by operator to adjust containment pressure to 0.50 +/- 0.15 psig) and receives a Containment Ventilation Isolation Signal (CVIS) to close.	OR13 – 3.599 scfh OR14 – 3.858 scfh OR15 – 4.634 scfh OR16 – 4.623 scfh OR17 – 4.348 scfh Acceptance Criteria: 7.4 scfh	Contingency Work Order pending LLRT results.
	COP-V1	Containment online purge supply isolation valve ORC-CIV 8" Posi Seal butterfly valve with Matryx pneumatic actuator	This valve may be open during power operation to provide filtered air for purging the containment (manually controlled by operator to adjust containment pressure to 0.50 +/- 0.15 psig) and receives a Containment Ventilation Isolation Signal (CVIS) to close.	OR13 – 3.085 scfh OR14 – 3.858 scfh OR15 – 4.634 scfh OR16 – 4.623 scfh OR17 – 4.348 scfh Acceptance Criteria: 7.4 scfh	Contingency Work Order pending LLRT results.

Valve Synopsis (continued)

Penetration	Valve	Description	Design Function	Service Life	Corrective Actions
X-35C	RC-FV2874	RCS Loop 1 sample valve ORC-CIV ½" Valcor Solenoid Valve	This valve is opened to obtain a sample and receives a "T" closure signal.	OR14 – 0.822 scfh OR15 – 1.328 scfh OR16 – 2.251 scfh OR17 – 0.650 scfh Acceptance Criteria: 147.8 scfh	There are no open corrective actions.
	RC-FV2894	RC Loop 1 sample valve ORC-CIV ½" Valcor Solenoid Valve	This valve is opened to obtain a sample and receives a "T" closure signal. This valve is utilized to obtain an RCS sample for boron concentration analysis to verify SDM during cold shutdown. If obtaining a sample is not possible, the operators verify adequate SDM by monitoring the volume of boric acid injected into the RCS.	OR14 – 0.822 scfh OR15 – 1.328 scfh OR16 – 2.251 scfh OR17 – 0.650 scfh Acceptance Criteria: 147.8 scfh	There are no open corrective actions.
	RC-FV2832	RC Loop 1 sample valve IRC-CIV ½" Valcor Solenoid Valve	This valve is opened to obtain a sample and receives a "T" closure signal. This valve is utilized to obtain an RCS sample for boron concentration analysis to verify SDM during cold shutdown. If obtaining a sample is not possible, the operators verify adequate SDM by monitoring the volume of boric acid injected into the RCS.	OR14 – 3.592 scfh OR15 – 4.083 scfh OR16 – 4.597 scfh OR17 – 3.900 scfh Acceptance Criteria: 147.8 scfh	There are no open corrective actions.
	RC-V314	RCS Loop 1 sample line containment penetration thermal relief valve IRC-CIV Crosby ¾" x 1", relief valve, OMNI Series	This valve is normally closed and opens to provide overpressure protection caused by thermal expansion of trapped fluid under accident conditions.	OR13 – 0.288 scfh OR14 – 0.288 scfh OR15 – 0.000 scfh OR16 – 0.816 scfh OR17 – 0.287 scfh Acceptance Criteria: 147.8 scfh	There are no open corrective actions.

Valve Synopsis (continued)

Penetration	Valve	Description	Design Function	Service Life	Corrective Actions
X-35D	RC-FV2876	RCS Loop 3 sample valve ORC-CIV ½" Valcor Solenoid Valve	This valve is opened to obtain a sample and receives a "T" closure signal.	OR14 – 0.615 scfh OR15 – 0.408 scfh OR16 – 0.707 scfh OR17 – 0.331 scfh Acceptance Criteria: 147.8 scfh	There are no open corrective actions.
	RC-FV2896	RC Loop 3 sample valve ORC-CIV ½" Valcor Solenoid Valve	This valve is opened to obtain a sample and receives a "T" closure signal. This valve is utilized to obtain an RCS sample for boron concentration analysis to verify SDM during cold shutdown. If obtaining a sample is not possible, the operators verify adequate SDM by monitoring the volume of boric acid injected into the RCS.	OR14 – 0.615 scfh OR15 – 0.408 scfh OR16 – 0.717 scfh OR17 – 0.331 scfh Acceptance Criteria: 147.8 scfh	There are no open corrective actions.
	RC-FV2833	RC Loop 3 sample valve IRC-CIV ½" Valcor Solenoid Valve	This valve is opened to obtain a sample and receives a "T" closure signal. This valve is utilized to obtain an RCS sample for boron concentration analysis to verify SDM during cold shutdown. If obtaining a sample is not possible, the operators verify adequate SDM by monitoring the volume of boric acid injected into the RCS.	OR14 – 0.513 scfh OR15 – 0.286 scfh OR16 – 0.000 scfh OR17 – 0.060 scfh Acceptance Criteria: 147.8 scfh	There are no open corrective actions.
	RC-V337	RCS Loop 3 sample line containment penetration thermal relief valve. IRC-CIV Lonergan ¾" x 1", relief valve, NJL40 Series	This valve is normally closed and opens to provide overpressure protection caused by thermal expansion of trapped fluid under accident conditions.	OR13 – 0.000 scfh OR14 – 0.000 scfh OR15 – 0.000 scfh OR16 – 0.287 scfh OR17 – 0.000 scfh Acceptance Criteria: 147.8 scfh	There are no open corrective actions.

Valve Synopsis (continued)

Penetration	Valve	Description	Design Function	Service Life	Corrective Actions
X-38A	FP-V588	Fire Protection water IRC-CIV 4" Velan swing check valve	This valve is normally closed in Modes 1-4, and has no active safety function.	OR13 – 14.843 scfh OR14 – 22.193 scfh OR15 – 9.556 scfh OR16 – 33.650 scfh OR17 – 11.100 scfh Acceptance Criteria: 147.8 scfh	There are no open corrective actions. Monitoring Flush of penetration following AF Testing for effectiveness.
	FP-V592	Containment Fire Protection water (hose stations) ORC-CIV 4" Velan manual gate valve	This valve is normally locked closed in Modes 1-4, and has no active safety function.	OR13 – 18.801 scfh OR14 – 12.155 scfh OR15 – 1.514 scfh OR16 – 21.184 scfh OR17 – 8.061 scfh Acceptance Criteria: 147.8 scfh	There are no open corrective actions. Monitoring Flush of penetration following AF Testing for effectiveness.
X-38B	CGC-V46	Containment Purge supply IRC-CIV 8" Velan manual gate valve	This check valve is normally closed and has no active safety function. The containment purge function is a defense in depth backup to the redundant- safety related hydrogen recombiners, and would be placed into service only if both recombiners failed or if the post LOCA hydrogen generation rate was significantly greater than the design basis generation rate. The purge subsystem relies on non-safety related systems such as service air, and is not required to function for SSD or design basis accident mitigation. This valve is subject to Appendix J Type C LLRT.	OR13 – 1.523 scfh OR14 – 2.584 scfh OR15 – 3.059 scfh OR16 – 2.265 scfh OR17 – 3.049 scfh Acceptance Criteria: 147.8 scfh	There are no open corrective actions.

Valve Synopsis (continued)

Penetration	Valve	Description	Design Function	Service Life	Corrective Actions
X-38B	CGC-V43	Containment Service Air supply ORC-CIV 2" Velan manual gate valve	This manual valve is normally closed and has no active safety function. The containment purge function is a defense in depth backup to the redundant- safety related hydrogen recombiners, and would be placed into service only if both recombiners failed or if the post LOCA hydrogen generation rate was significantly greater than the design basis generation rate. The purge subsystem relies on non-safety related systems such as service air, and is not required to function for SSD or design basis accident mitigation. This valve is subject to Appendix J Type C LLRT.	OR13 – 0.812 scfh OR14 – 1.443 scfh OR15 – 1.525 scfh OR16 – 0.512 scfh OR17 – 0.993 scfh Acceptance Criteria: 147.8 scfh	There are no open corrective actions.
	CGC-V44	Containment Service Air supply ORC-CIV 2" Velan manual gate valve	This manual valve is normally closed and has no active safety function. The containment purge function is a defense in depth backup to the redundant- safety related hydrogen recombiners, and would be placed into service only if both recombiners failed or if the post LOCA hydrogen generation rate was significantly greater than the design basis generation rate. The purge subsystem relies on non-safety related systems such as service air, and is not required to function for SSD or design basis accident mitigation. This valve is subject to Appendix J Type C LLRT.	OR13 – 0.812 scfh OR14 – 1.443 scfh OR15 – 1.525 scfh OR16 – 0.512 scfh OR17 – 0.993 scfh Acceptance Criteria: 147.8 scfh	There are no open corrective actions.

Valve Synopsis (continued)

Penetration	Valve	Description	Design Function	Service Life	Corrective Actions
X-38B	CGC-V45	Containment alternate Purge (portable air compressor) supply ORC-CIV 8" Velan manual gate valve	This manual valve is normally closed and has no active safety function. The containment purge function is a defense in depth backup to the redundant- safety related hydrogen recombiners, and would be placed into service only if both recombiners failed or if the post LOCA hydrogen generation rate was significantly greater than the design basis generation rate. The purge subsystem relies on non-safety related systems such as service air, and is not required to function for SSD or design basis accident mitigation. This valve is subject to Appendix J Type C LLRT.	OR13 – 0.812 scfh OR14 – 1.443 scfh OR15 – 1.525 scfh OR16 – 0.512 scfh OR17 – 0.993 scfh Acceptance Criteria: 147.8 scfh	There are no open corrective actions.

3. Provide an explanation of why these six penetrations have not qualified for an extended test interval.

   The six penetrations not on extended frequency are:

   Restricted by Regulatory Guide 1.163

   - X-16, valves COP-V3 and COP-V4
     - Regulatory Guide 1.163 restricts these containment purge and vent valves to 30 months. Seabrook tests these penetrations every refueling outage (e.g., 18 months) so as to not exceed the 30 month restriction.
   - X-18, valves COP-V2 and COP-V1
     - Regulatory Guide 1.163 restricts these containment purge and vent valves to 30 months. Seabrook tests these penetrations every refueling outage (e.g., 18 months) so as to not exceed the 30 month restriction.

   Restricted by Site

   - X-35C, valves RC-FV2874 and RC-FV2894/RC-FV2832 and RC-V314
     - The LLRT is used to satisfy the 2 year Inservice Test position indication test requirement for one of the Valcor solenoid valves. If another type of IST position indication test is performed to address the IST requirement, then the LLRT for this penetration could be extended.
   - X-35D, valves RC-FV2876 and RC-FV2896/RC-FV2833 and RC-V337
     - The LLRT is used to satisfy the 2 year Inservice Test position indication test requirement for one of the Valcor solenoid valves. If another type of IST position indication test is performed to address the IST requirement, then the LLRT for this penetration could be extended.
   - X-38A, valves FP-V588/FP-V592
     - The LLRT is currently done every outage due to system cleanliness issues that caused higher than desired AS Found test results. A system flush is then performed and had been effective in reducing the leakage. This sequence will be monitored over several cycles to see if it is effective in maintaining the penetration leakage at a low repeatable value.
   - X-38B, valves CGC-V46/CGC-V43 and CGC-V44 and CGC-V45
     - opened for containment services during outages – steam generator air supply line. As such As Found and As left testing is done.

   The valve performance of for these penetrations (X-35C, X-35D, X-38A, and X-38B) are suitable to justify placing on extended intervals when the trended supports the extension or when other tests are written to credit the inservice testing position indication tests, or when the penetrations are no longer required to support other refueling outage functions.

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.

NextEra Response to SBPB-RAI-4

The correct references should be Sections 3.2.1 through 3.2.7. The applicable sentence should read "Reference Sections 3.2.1 through 3.2.7"

Enclosure 2 to SBK-L-16165

Changes to Attachment 4, "Risk Impact Assessment" from NextEra Energy Seabrook, LLC letter SBK-L-16029, "License Amendment Request 16-01, Request to Extend Containment Leakage Test Frequency, March 31, 2016 (ML16095A278 and ML16159A194)

Table 5-2: Seabrook Station Categorized Accident Classes and Frequencies			
EPRI Class	**Seabrook Station Release Category**	**Frequency Based on Categorized Results (per yr)**	**Adjusted Frequency Using Scale Factor of 1.0 (per yr)**
7	Late Containment Failure (LATE) and Containment Failure (~~SE02, SE03~~, LE04)	3.14E-06	3.14E-06
8	Containment Bypass (SE02 and LE02) and SGTR (SE01 and LE01)	6.46E-07	6.46E-07
Total Frequency		1.20E-05	1.20E-05

Class 1 Sequences. This group consists of all core damage accident progression bins for which the containment remains intact (modeled as Technical Specification Leakage). The frequency per year is initially determined from the Containment Intact Level 2 Release Category listed in Table 4-5 minus the EPRI Class 3a and 3b frequency, which are calculated below.

Class 2 Sequences. This group consists of all core damage accident progression bins for which a failure to isolate the containment occurs. The frequency per year for these sequences is obtained from the Containment Isolation Failures listed in Table 5-2.

Class 3 Sequences. This group consists of all core damage accident progression bins for which a pre-existing leakage in the containment structure (e.g., containment liner) exists. The containment leakage for these sequences can be either small (in excess of design allowable but <10La) or large (>100La).

The respective frequencies per year are determined as follows:

$PROB_{class\_3a}$ = probability of small pre-existing containment liner leakage

= 0.0092 [see Section 4.3]

$PROB_{class\_3b}$ = probability of large pre-existing containment liner leakage

= 0.0023 [see Section 4.3]

As described in Section 4.3, additional consideration is made to not apply these failure probabilities on those cases that are already LERF scenarios (i.e., the Class 2 and Class 8 contributions).

Class 3a Frequency = 0.0092 * (CDF - (Class 2 + Class 8))

= 0.0092 * (1.20E-05/yr – (2.31E-07/yr + 6.46E-07/yr)) = 1.02E-07/yr

Class 3b Frequency = 0.0023 * (CDF - (Class 2 + Class 8))

Table 5-3: Radionuclide Release Frequencies as a Function of Accident Class (Seabrook Station Base Case)			
Accident Classes (Containment Release Type)	Description	Frequency (per Rx-yr)	
		EPRI Methodology	EPRI Methodology Plus Corrosion[1]
1	No Containment Failure	7.82E-06	7.82E-06
2	Large Isolation Failures (Failure to Close)	2.31E-07	2.31E-07
3a	Small Isolation Failures (liner breach)	1.02E-07	1.02E-07
3b	Large Isolation Failures (liner breach)	2.55E-08	2.56E-08
4	Small Isolation Failures (Failure to seal –Type B)	N/A	N/A
5	Small Isolation Failures (Failure to seal—Type C)	N/A	N/A
6	Other Isolation Failures (e.g., dependent failures)	N/A	N/A
7	Failures Induced by Phenomena (Early and Late)	3.14E-06	3.14E-06
8	Bypass (Interfacing System LOCA)	6.46E-07	6.46E-07
CDF	All CET end states	1.20E-05	1.20E-05

1. Note that this is based on data developed in Section 4.4. Only Classes 1 and 3b are impacted by the corrosion.

5.2 Step 2 - Develop Plant Specific Person-Rem Dose (Population Dose) Per Reactor Year

Plant specific release analyses were performed to estimate the person-rem doses to the population within a 50 mile radius from the plant, and summarized in Table 4-4. The results of applying these releases to the EPRI containment failure classification are as follows:

Class 1 = 2.77E+03 person-rem (Note 1)

Class 2 = 1.44E+06 person rem (Note 2)

Class 3a = 2.77E+03 person-rem x 10La = 2.77E+04 person-rem (Note 3)

Class 3b = 2.77E+03 person-rem x 100La = 2.77E+05 person-rem (Note 3)

Class 4 = Not analyzed

Class 5 = Not analyzed

Class 6 = Not analyzed

Class 7 = ~~9.25E+06~~ 9.83E+06 person rem (Note 4)

Class 8 = ~~6.46E+05~~ 2.63E+06 person-rem (Note 5)

Notes:

(1) Class 1 is assigned the dose from the frequency weighted average of the dose from release categories Intact1 and Intact2 from Table 4-2 and Table 4-4.

(2) Class 2 is assigned the dose from the frequency weighted average of the dose from release categories LE3 and SE3 from Table 4-2 and Table 4-4.

(3) The Class 3a and 3b dose are related to the Class 1 leakage rate as shown. While no pre-existing leakage in excess of 21 La has been identified for any historical ILRT event, Class 3b releases are conservatively assessed at 100La. Class 3a releases are conservatively assessed at 10La. This is consistent with the guidance provided in EPRI Report No. 1009325, Revision 2-A.

(4) Class 7 is assigned the frequency weighted average of the dose from release categories LE4, LL3, LL4, LL5, and SELL from Table 4-4.

(5) Class 8 sequences involve containment bypass failures; as a result, the person-rem dose is not based on normal containment leakage. The releases for this class are assigned from the frequency weighted average of the dose from release categories SE1, SE2, LE1 and LE2 from Table 4-4.

In summary, the population dose estimates derived for use in the risk evaluation per the EPRI methodology (Reference 2) containment failure classifications, and consistent with the NEI guidance (Reference 3) as modified by EPRI Report No. 1009325, Revision 2-A are provided in Table 5-4.

Table 5-4: Seabrook Station Population Dose Estimates for Population Within 50 Miles

Accident Classes (Containment Release Type)	Description	Person-Rem (50 miles)
1	No Containment Failure	2.77E+03
2	Large Isolation Failures (Failure to Close)	1.44E+06
3a	Small Isolation Failures (liner breach)	2.77E+04
3b	Large Isolation Failures (liner breach)	2.77E+05
4	Small Isolation Failures (Failure to seal-Type B)	N/A
5	Small Isolation Failures (Failure to seal-Type C)	N/A
6	Other Isolation Failures (e.g., dependent failures)	N/A
7	Failures Induced by Phenomena (Early and Late)	~~9.25E+06~~ 9.83E+06
8	Bypass (Interfacing System LOCA and Unisolated SGTR)	~~6.46E+05~~ 2.63E+06

Table 5-5: Seabrook Station Annual Dose as a Function of Accident Class; Characteristic of Conditions for ILRT Required 3/10 Years							
Accident Classes (Cnmt Release Type)	**Description**	**Person-Rem (50 miles)**	**EPRI Methodology**		**EPRI Methodology Plus Corrosion**		**Change Due to Corrosion Person-Rem/yr(1)**
			Frequency (per Rx-yr)	**Person-Rem/yr (50 miles)**	**Frequency (per Rx-yr)**	**Person- Rem/yr (50 miles)**	
1	No Containment Failure (2)	2.77E+03	7.82E-06	2.16E-02	7.82E-06	2.16E-02	-2.91E-07
2	Large Isolation Failures (Failure to Close)	1.44E+06	2.31E-07	3.34E-01	2.31E-07	3.34E-01	0.00E+00
3a	Small Isolation Failures (liner breach)	2.77E+04	1.02E-07	2.82E-03	1.02E-07	2.82E-03	0.00E+00
3b	Large Isolation Failures (liner breach)	2.77E+05	2.55E-08	7.06E-03	2.56E-08	7.09E-03	2.91E-05
4	Small Isolation Failures (Failure to seal -Type B)	N/A	N/A	N/A	N/A	N/A	N/A
5	Small Isolation Failures (Failure to seal-Type C)	N/A	N/A	N/A	N/A	N/A	N/A
6	Other Isolation Failures (e.g., dependent failures)	N/A	N/A	N/A	N/A	N/A	N/A
7	Failures Induced by Phenomena (Early and Late)	~~9.25E+06~~ 9.83E+06	3.14E-06	~~2.91E+01~~ 3.09E+01	3.14E-06	~~2.91E+01~~ 3.09E+01	0.00E+00

Table 5-5: Seabrook Station Annual Dose as a Function of Accident Class; Characteristic of Conditions for ILRT Required 3/10 Years							
Accident Classes (Cnmt Release Type)	**Description**	**Person-Rem (50 miles)**	**EPRI Methodology**		**EPRI Methodology Plus Corrosion**		**Change Due to Corrosion Person-Rem/yr(1)**
			Frequency (per Rx-yr)	**Person-Rem/yr (50 miles)**	**Frequency (per Rx-yr)**	**Person- Rem/yr (50 miles)**	
8	Bypass (Interfacing System LOCA)	~~6.46E+05~~ 2.63E+06	6.46E-07	~~4.18E-01~~ 1.70E+00	6.46E-07	~~4.18E-01~~ 1.70E+00	0.00E+00
CDF	All CET end states	N/A	1.20E-05	~~2.99E+01~~ 3.29E+01	1.20E-05	~~2.99E+01~~ 3.29E+01	2.88E-05

1) Only release Classes 1 and 3b are affected by the corrosion analysis.
2) Characterized as 1La release magnitude consistent with the derivation of the ILRT non-detection failure probability for ILRTs. Release classes 3a and 3b include failures of containment to meet the Technical Specification leak rate.

Table 5-6: Seabrook Station Annual Dose as a Function of Accident Class; Characteristic of Conditions for ILRT Required 1/10 Years							
Accident Classes (Cnmt Release Type)	**Description**	**Person-Rem (50 miles)**	**EPRI Methodology**		**EPRI Methodology Plus Corrosion**		**Change Due to Corrosion Person-Rem/yr[(1)]**
			Frequency (per Rx-yr)	**Person-Rem/yr (50 miles)**	**Frequency (per Rx-yr)**	**Person-Rem/yr (50 miles)**	
1	No Containment Failure [(2)]	2.77E+03	7.52E-06	2.08E-02	7.52E-06	2.08E-02	-9.69E-07
2	Large Isolation Failures (Failure to Close)	1.44E+06	2.31E-07	3.34E-01	2.31E-07	3.34E-01	0.00E+00
3a	Small Isolation Failures (liner breach)	2.77E+04	3.40E-07	9.41E-03	3.40E-07	9.41E-03	0.00E+00
3b	Large Isolation Failures (liner breach)	2.77E+05	8.49E-08	2.35E-02	8.53E-08	2.36E-02	9.69E-05
4	Small Isolation Failures(Failure to seal-Type B)	N/A	N/A	N/A	N/A	N/A	N/A
5	Small Isolation Failures (Failure to seal-Type C)	N/A	N/A	N/A	N/A	N/A	N/A
6	Other Isolation Failures (e.g., dependent failures)	N/A	N/A	N/A	N/A	N/A	N/A
7	Failures Induced by Phenomena (Early and Late)	~~9.25E+06~~ 9.83E+06	3.14E-06	~~2.91E+01~~ 3.09E+01	3.14E-06	~~2.91E+01~~ 3.09E+01	0.00E+00

Table 5-6: Seabrook Station Annual Dose as a Function of Accident Class; Characteristic of Conditions for ILRT Required 1/10 Years							
Accident Classes (Cnmt Release Type)	**Description**	**Person-Rem (50 miles)**	**EPRI Methodology**		**EPRI Methodology Plus Corrosion**		**Change Due to Corrosion Person-Rem/yr(1)**
			Frequency (per Rx-yr)	**Person-Rem/yr (50 miles)**	**Frequency (per Rx-yr)**	**Person-Rem/yr (50 miles)**	
8	Bypass (Interfacing System LOCA)	~~6.46E+05~~ 2.63E+06	6.46E-07	~~4.18E-01~~ 1.70E+00	6.46E-07	~~4.18E-01~~ 1.70E+00	0.00E+00
CDF	All CET end states	N/A	1.20E-05	~~2.99E+01~~ 3.30E+01	1.20E-05	~~2.99E+01~~ 3.30E+01	9.59E-05

1) Only release Classes 1 and 3b are affected by the corrosion analysis.
2) Characterized as 1La release magnitude consistent with the derivation of the ILRT non-detection failure probability for ILRTs. Release classes 3a and 3b include failures of containment to meet the Technical Specification leak rate.

Table 5-7: Seabrook Station Annual Dose as a Function of Accident Class; Characteristic of Conditions for ILRT Required 1/15 Years							
Accident Classes (Cnmt Release Type)	**Description**	**Person-Rem (50 miles)**	**EPRI Methodology**		**EPRI Methodology Plus Corrosion**		**Change Due to Corrosion Person-Rem/yr(1)**
			Frequency (per Rx-yr)	**Person-Rem/yr (50 miles)**	**Frequency (per Rx-yr)**	**Person-Rem/yr (50 miles)**	
7	Failures Induced by Phenomena (Early and Late)	~~9.25E+06~~ 9.83E+06	3.14E-06	~~2.91E+01~~ 3.09E+01	3.14E-06	~~2.91E+01~~ 3.09E+01	0.00E+00
8	Bypass (Interfacing System LOCA)	~~6.46E+05~~ 2.63E+06	6.46E-07	~~4.18E-01~~ 1.70E+01	6.46E-07	~~4.18E-01~~ 1.70E+01	0.00E+00
CDF	All CET end states	N/A	1.20E-05	~~2.99E+01~~ 3.30E+01	1.20E-05	~~2.99E+01~~ 3.30E+01	1.44E-04

1) Only release Classes 1 and 3b are affected by the corrosion analysis.
2) Characterized as 1La release magnitude consistent with the derivation of the ILRT non-detection failure probability for ILRTs. Release classes 3a and 3b include failures of containment to meet the Technical Specification leak rate.

Table 5-8: Seabrook Station ILRT Cases: Base, 3 to 10, and 3 to 15 Yr Extensions (Including Age Adjusted Steel Liner Corrosion Likelihood)

EPRI Class	DOSE Per-Rem	Base Case 3 in 10 Years		Extend to 1 in 10 Years		Extend to 1 in 15 Years	
		CDF/Yr	Per-Rem/Yr	CDF/Yr	Per-Rem/Yr	CDF/Yr	Per-Rem/Yr
1	2.77E+03	7.82E-06	2.16E-02	7.52E-06	2.08E-02	7.31E-06	2.02E-02
2	1.44E+06	2.31E-07	3.34E-01	2.31E-07	3.34E-01	2.31E-07	3.34E-01
3a	2.77E+04	1.02E-07	2.82E-03	3.40E-07	9.41E-03	5.10E-07	1.41E-02
3b	2.77E+05	2.56E-08	7.09E-03	8.53E-08	2.36E-02	1.28E-07	3.55E-02
7	~~9.25E+06~~ 9.83E+06	3.14E-06	~~2.91E+01~~ 3.09E+01	3.14E-06	~~2.91E+01~~ 3.09E+01	3.14E-06	~~2.91E+01~~ 3.09E+01
8	~~6.46E+05~~ 2.63E+06	6.46E-07	~~4.18E-01~~ 1.70E+00	6.46E-07	~~4.18E-01~~ 1.70E+00	6.46E-07	~~4.18E-01~~ 1.70E+00
Total	N/A	1.20E-05	~~2.99E+01~~ 3.29E+01	1.20E-05	~~2.99E+01~~ 3.30E+01	1.20E-05	~~2.99E+01~~ 3.30E+01

		Base Case 3 in 10 Years	Extend to 1 in 10 Years	Extend to 1 in 15 Years
ILRT Dose Rate from 3a and 3b Per-Rem/Yr		9.92E-03	3.30E-02	4.96E-02
Delta Total Dose Rate[1]	From 3 yr	N/A	2.23E-02	3.82E-02
	From 10 yr	N/A	N/A	1.60E-02
% change in dose rate from base	From 3 yr	N/A	0.07%	~~0.13%~~ 0.12%
	From 10 yr	N/A	N/A	0.05%
3b Frequency (LERF) /Yr		2.56E-08	8.53E-08	1.28E-07
Delta LERF	From 3 yr	N/A	5.97E-08	1.02E-07
	From 10 yr	N/A	N/A	4.28E-08
CCFP %		33.80%	34.30%	34.65%
Delta CCFP %	From 3 yr	N/A	0.50%	0.86%
	From 10 yr	N/A	N/A	0.36%

[1] The overall difference in total dose rate is less than the difference of only the 3a and 3b categories between two testing intervals. This is because the overall total dose rate includes contributions from other categories that do not change as a function of time, e.g., the EPRI Class 2 and 8 categories, and also due to the fact that the Class 1 person-rem/yr decreases when extending the ILRT frequency.

The ILRT test is focused on performing a periodic validation of the containment liner leak tightness, which is a condition that is independent of fire events risk. Therefore the ILRT extension has no direct effect on the core damage from fire events.

As shown above, fire events do not contribute significantly to LERF (1.3E-10/yr). This is consistent with Seabrook's relatively low total LERF contribution (1.5E-07/yr) from all internal and external events, with the total LERF of all events being approximately two orders of magnitude lower than total CDF (1.2E-05/yr). The relatively low total LERF is a reflection of Seabrook's robust containment design and associated release mitigation capability. Because Seabrook's baseline LERF is relatively low, the "added" 3b-LERF introduced by the ILRT extension, although very small, is relatively large compared to the low baseline LERF. Thus, the delta-LERF as a result of applying the industry ILRT methodology appears as a relatively large increase for Seabrook.

As noted in Appendix A the Seabrook fire PRA is in the process of being formally upgraded to meet the latest industry methods (guidance in NUREG/CR-6850 (Reference 33) and CC II of the ASME PRA Standard. Insights gained thus far in the update process continue to suggest that the current fire PRA framework is comprehensive and a realistic representation of fire risk. However, as a result of applying the latest industry fire risk methods, it is recognized that the final updated fire-induced CDF/LERF could increase above the current values. Therefore, a fire/ILRT risk sensitivity evaluation was performed by conservatively assuming that the current fire-induced CDF is increased by a factor of 2. This sensitivity updates the calculation of core damage frequency in the analysis shown in Table 4-1, and increases the fire contribution to the release category frequencies in Table 4-2 by a factor of 2. All other calculations in the assessment are unchanged. The results of this evaluation are shown in Table 6-2. This evaluation shows that the 15 year ILRT test interval 3b LERF would become 1.45E-07/yr resulting in a delta-LERF of 1.16E-07/yr. This represents an increase in the delta-LERF of approximately 1.4E-08/yr above the baseline delta-LERF of 1.02E-07/yr. This shows that the baseline delta-LERF is relatively insensitive to the assumed increase in fire risk.

Conclusion - The baseline quantitative fire CDF and LERF contributions are judged appropriate for use in the delta LERF impact of the ILRT Type A test interval extension. No additional fire risk insights were identified for further consideration.

Table 6-2: Senstivity to Doubling of Fire CDF for Seabrook Station

EPRI Class	DOSE Per-Rem	Base Case 3 in 10 Years		Extend to 1 in 10 Years		Extend to 1 in 15 Years	
		CDF/Yr	Per-Rem/Yr	CDF/Yr	Per-Rem/Yr	CDF/Yr	Per-Rem/Yr
1	2.77E+03	8.97E-06	2.48E-02	8.64E-06	2.39E-02	8.39E-06	2.32E-02
2	1.45E+06	2.36E-07	3.44E-01	2.36E-07	3.44E-01	2.36E-07	3.44E-01
3a	2.77E+04	1.16E-07	3.20E-03	3.85E-07	1.07E-02	5.78E-07	1.60E-02
3b	2.77E+05	2.90E-08	8.03E-03	9.66E-08	2.68E-02	1.45E-07	4.02E-02
7	~~9.16E+06~~ 9.69E+06	3.44E-06	~~3.15E+01~~ 3.34E+01	3.44E-06	~~3.15E+01~~ 3.34E+01	3.44E-06	~~3.15E+01~~ 3.34E+01

Table 6-2: Senstivity to Doubling of Fire CDF for Seabrook Station							
8	~~6.46E+05~~ 2.63E+06	6.46E-07	~~4.18E-01~~ 1.70E+00	6.46E-07	~~4.18E-01~~ 1.70E+00	6.46E-07	~~4.18E-01~~ 1.70E+00
Total	N/A	1.34E-05	~~3.23E+01~~ 3.55E+01	1.34E-05	~~3.24E+01~~ 3.55E+01	1.34E-05	~~3.24E+01~~ 3.55E+01
ILRT Dose Rate from 3a and 3b		1.12E-02		3.74E-02		5.62E-02	
Delta Total Dose Rate	From 3 yr	N/A		2.52E-02		4.33E-02	
	From 10 yr	N/A		N/A		1.81E-02	
% change in dose rate from base	From 3 yr	N/A		~~0.08%~~ 0.07%		~~0.13%~~ 0.12%	
	From 10 yr	N/A		N/A		~~0.06%~~ 0.05%	
3b Frequency (LERF)		2.90E-08		9.66E-08		1.45E-07	
Delta LERF	From 3 yr	N/A		6.76E-08		1.16E-07	
	From 10 yr	N/A		N/A		4.85E-08	
CCFP %		32.40%		32.90%		33.26%	
Delta CCFP%	From 3 yr	N/A		0.50%		0.86%	
	From 10 yr	N/A		N/A		0.36%	

Seismic Events - Qualitative Risk Impacts on ILRT Extension

The Seabrook Seismic hazards PRA includes a quantitative assessment of at-power seismic risk. In the base case plant risk model, seismic events contribute approximately 27% to the total at-power CDF. The seismic hazards PRA is based on the probabilistic hazard curves in Reference 30. The base case quantitative CDF/LERF results for seismic events are as follows:

CDF: 3.25E-06/yr

LERF: 9.85E-08/yr

The proposed extension of the ILRT interval does not impact the frequency of any seismic initiating event nor does the ILRT extension impact the reliability of active equipment credited in seismic initiating event sequences. The Seabrook plant is a relatively late vintage design and has a robust structural seismic capacity. Seismic risk is dominated by large, beyond-design-basis seismic events that result in transients with loss of offsite power with failure of the Emergency Diesel Generators and Supplemental Emergency Power Supply diesels (station blackout condition) and failure of other non-seismically supported and seismically supported equipment needed for core cooling. Large seismic events can also result in large LOCA events with seismic-induced failure of ECCS equipment. Smaller size seismic events are less likely to cause a large LOCA or blackout-type event and support systems and ECCS are more likely to remain

Table 6-3: Senstivity to GI-199 Seismic CDF for Seabrook Station							
EPRI Class	DOSE Per-Rem	Base Case 3 in 10 Years		Extend to 1 in 10 Years		Extend to 1 in 15 Years	
		CDF/Yr	Per-Rem/Yr	CDF/Yr	Per-Rem/Yr	CDF/Yr	Per-Rem/Yr
1	2.77E+03	1.08E-05	3.00E-02	1.04E-05	2.87E-02	1.00E-05	2.78E-02
2	1.41E+06	5.81E-07	8.17E-01	5.81E-07	8.17E-01	5.81E-07	8.17E-01
3a	2.77E+04	1.60E-07	4.44E-03	5.34E-07	1.48E-02	8.02E-07	2.22E-02
3b	2.77E+05	4.02E-08	1.11E-02	1.34E-07	3.71E-02	2.01E-07	5.57E-02
7	~~9.77E+06~~ 1.05E+07	6.40E-06	~~6.25E+01~~ 6.73E+01	6.40E-06	~~6.25E+01~~ 6.73E+01	6.40E-06	~~6.25E+01~~ 6.73E+01
8	~~6.41E+05~~ 2.80E+06	6.68E-07	~~4.28E-01~~ 1.87E+00	6.68E-07	~~4.28E-01~~ 1.87E+00	6.68E-07	~~4.28E-01~~ 1.87E+00
Total	N/A	1.87E-05	~~6.38E+01~~ 7.01E+01	1.87E-05	~~6.38E+01~~ 7.01E+01	1.87E-05	~~6.38E+01~~ 7.01E+01
ILRT Dose Rate from 3a and 3b		1.56E-02		5.19E-02		7.79E-02	
Delta Total Dose Rate	From 3 yr	N/A		3.50E-02		6.01E-02	
	From 10 yr	N/A		N/A		2.51E-02	
% change in dose rate from base	From 3 yr	N/A		0.05%		0.09%	
	From 10 yr	N/A		N/A		0.04%	
3b Frequency (LERF)		4.02E-08		1.34E-07		2.01E-07	
Delta LERF	From 3 yr	N/A		9.38E-08		1.61E-07	
	From 10 yr	N/A		N/A		6.72E-08	
CCFP %		41.15%		41.66%		42.02%	
Delta CCFP%	From 3 yr	N/A		0.50%		0.86%	
	From 10 yr	N/A		N/A		0.36%	

Other External Events - Qualitative Risk Impacts on ILRT Extension

Other external events include LOSP due to severe weather, external flooding, high winds, tornado, transportation and near-by facility hazards, turbine missile, etc. Some of these events (e.g., weather-related LOSP) are quantified in the internal events PRA model while others are screened out based on the low probability of occurrence and rigorous plant design features. Collectively, all "other" external events are judged to have a very small contribution to CDF/LERF. The proposed extension of the ILRT test interval does not impact the initiating event frequencies of other external events nor does the ILRT extension impact the reliability of active equipment needed to mitigate these events. The ILRT test is focused on performing a periodic validation of the containment liner leak tightness, which is a condition that is independent of external event risk. Therefore the ILRT extension has no direct effect on the core damage and release mitigation capability from external events.