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

January 5, 2011

Mr. Paul Freeman
Site Vice President
c/o Mr. Michael O'Keefe
NextEra Energy Seabrook, LLC
P.O. Box 300
Seabrook, NH 03874

SUBJECT: REQUEST FOR ADDITIONAL INFORMATION RELATED TO THE REVIEW OF THE SEABROOK STATION LICENSE RENEWAL APPLICATION (TAC NO. ME4028)

Dear Mr. Freeman:

By letter dated May 25, 2010, NextEra Energy Seabrook, LLC, submitted an application pursuant to Title 10 of the *Code of Federal Regulations* Part 54, to renew Operating License NPF-86 for Seabrook Station, Unit 1, for review by the U.S. Nuclear Regulatory Commission (NRC or the staff). The staff is reviewing the information contained in the license renewal application and has identified, in the enclosure, areas where additional information is needed to complete the review.

The request for additional information was discussed with Mr. Rick Cliche, and a mutually agreeable date for the response is within 30 days from the date of this letter. If you have any questions, please contact me at 301-415-1427 or by e-mail at richard.plasse@nrc.gov.

Sincerely,

Richard Plasse, Project Manager
Projects Branch 2
Division of License Renewal
Office of Nuclear Reactor Regulation

Docket No. 50-443

Enclosure:
As stated

cc w/encl: Distribution via Listserv

Seabrook Station
License Renewal Application
Request for Additional Information Set 7
Time-Limited Aging Analysis

RAI 4.7.4-1

Background
License renewal application (LRA) Section 4.7.4 provides the applicant's basis for dispositioning the cumulative usage factor (CUF) analyses for Class 1 High Energy Line Break (HELB) locations in accordance with 10 CFR 54.21(c)(1)(i). The analyses will remain valid for the period of extended operation. LRA Section 4.7.4 refers to the design analyses for these piping locations, as discussed in Updated Final Safety Analysis Report (UFSAR) Section 3.6(B), "Protection Against Dynamic Effects Associated with the Postulated Rupture of Piping."

UFSAR Section 3.6(B) discusses eight primary reactor coolant loop locations that were approved in accordance with the applicant's leak-before-break (LBB) analysis. The applicant's LBB analysis permitted the removal of dynamic effect considerations from the scope of the applicant's large break loss of coolant accident (LOCA) analysis, provided in Chapter 15 of the UFSAR. The U.S. Nuclear Regulatory Commission's (NRC or the staff) approval of the LBB analysis for these piping locations was issued in accordance with 10 CFR Part 50, Appendix A, General Design Criteria, Criterion 4, *Dynamic Effects*.

Issue
The LRA does not identify which ASME Code Class 1 reactor coolant pressure boundary locations in UFSAR Section 3.6(B) are within the scope of the LBB analysis and which of these piping locations currently remain within the scope of the applicant's HELB CUF analyses.

LRA Section 4.7.4 also dispositioned the CUF analyses in accordance with 10 CFR 54.21(c)(1)(i) without identifying the current CUF values of record or the design basis transients in LRA Table 4.3.1-2 that are applicable to the HELB locations. The staff is not able to determine whether the disposition of 10 CFR 54.21(c)(1)(i) for each of these piping locations is supported by the design transients projection basis that was provided in LRA Table 4.3.1-2.

Request
1. Identify the ASME Code Class 1 piping locations discussed in UFSAR Section 3.6(B) that are within the scope of the LBB analysis and LRA Section 4.7.3. Identify the ASME Code Class 1 piping locations discussed in UFSAR Section 3.6(B) that are within the scope of the CUF analyses that are discussed in LRA Section 4.7.4. Clarify whether the current design basis uses the LBB analysis to replace any of the original CUF analyses for HELB piping locations.

2. Provide the CUF value of record and the design basis transients that are applicable to each HELB piping location that are associated with LRA Section 4.7.4. Provide the design cycle limits and 60-year projected cycle for the applicable transients if the projections have not been included in LRA Table 4.3.1-3.

RAI 4.7.14-1

Background
LRA Section 4.7.14 summarizes the plant-specific time-limited aging analysis (TLAA) for environmental qualification (EQ) of the Emergency Diesel Generator (EDG). The applicant stated that the evaluation was performed by the original EDG manufacturer to support EQ of the EDGs in accordance with IEEE Standard 323. The applicant dispositioned this TLAA in accordance with 10 CFR 54.21(c)(1)(i), the analyses will remain valid for the period of extended operation. LRA Section 4.7.14 discusses the number of projected EDG scheduled start and unscheduled start cycles compared to the 5454 cycles that was used for the EQ evaluation. Specifically, the applicant indicated that the EQ was based on a total of 5454 full temperature cycles of EDG operation. The applicant also stated that the estimated number of cycles through 60 years of licensed operations is 2160 and that this accounts for EDG maintenance activities, EDG testing activities, and EDG starts during postulated design basis transient and accident events.

10 CFR 54.3 identifies six criteria that must be met for an analysis to be defined as a TLAA. Two of these criteria are: (1) Involve systems, structures, and components within the scope of license renewal, as delineated in § 54.4(a); and (2) Consider the effects of aging. For an analysis in the current licensing basis (CLB) that meets the definition of a TLAA, the applicant may disposition it in accordance with 10 CFR 54.21(c)(1)(i) if it is demonstrated that the analyses will remain valid for the period of extended operation.

Issue 1
LRA Section 4.7.14 does not identify which of these components were analyzed in the TLAA's cycle-dependent EQ analysis. LRA Section 4.7.14 also does not discuss the aging effect(s) in IEEE Standard 323 that were evaluated in the EDG EQ analysis.

Request 1
Summarize the aging effects within the IEEE Standard 323, and identify the aging effects that were analyzed in the EQ analysis for these components. Clarify how the number of analyzed cycles (5454 cycles) in the EQ analysis is associated with the aging effects and any applicable acceptance criteria for these aging effects.

Issue 2
It is not clear to the staff which design transients (i.e., those listed in LRA Table 4.3.1-2 or additional transients not listed in LRA Table 4.3.1-2) would result in a scheduled or unscheduled start of the EDGs. The staff requires identification of all transients or activities that will initiate a start of the EDGs and the 60-year projections for these transients so a comparison can be made to the 5454 cycle limit on EDG, to determine if the disposition of 10 CFR 54.21(c)(1)(i) is appropriate.

It is also not clear to the staff when the applicant compares the number of analyzed cycles to the projections of the applicable transients or activities that will initiate an EDG start, if this is performed individually or cumulatively.

Request 2
Identify all transients in LRA Table 4.3.1-2 that will result in a scheduled or unscheduled start of the EDGs.

Identify any additional transients, beyond those that are listed in LRA Table 4.3.1-2, that can result in a scheduled or unscheduled start of the EDG and provide the 60-year projections, consistent with LRA Table 4.3.1-3.

Clarify whether the 60-year projection of the applicable EDG scheduled and unscheduled start transients and activities is performed on a cumulative or individual transient projection basis. If it is the latter basis, justify why the analysis did not perform the projections using a separate acceptance limit for each of the transient or activity that could result in a scheduled or an unscheduled start of the EDGs.

RAI 4.3-1

Background
LRA Section 4.3 states that the metal fatigue TLAAs that are evaluated in the LRA fall into the following three categories:

1. Explicit fatigue analyses for nuclear steam supply system (NSSS) pressure vessels and components prepared in accordance with ASME Section III, Class A or Class 1 rules developed as part of the original design.

2. Supplemental explicit fatigue analyses for piping and components that were prepared in accordance with ASME Section III rules to evaluate transients that were identified after the original design analyses were completed, such as pressurizer surge line thermal stratification, and also include reactor vessel internal component fatigue analyses.

3. New fatigue analyses (also in accordance with ASME Section III, Class 1 rules) prepared for license renewal to evaluate the effects of the reactor water environment on the sample of high fatigue locations applicable to newer vintage Westinghouse Plants, as identified in Section 5.5 of NUREG/CR-6260, and using the methodology presented in LRA Section 4.3.4.

LRA Section 4.3.1 states that the most limiting numbers of transients used in these component analyses are given in LRA Table 4.3.1-2 (also listed in FSAR Table 3.9(N)-1), and those numbers are considered to be design limits.

Issue
The LRA does not provide a list of reactor coolant system (RCS) locations for which fatigue CUF analyses were performed in the CLB, or the CUF value of record for these locations. In particular, the LRA does not provide any CUF values for the NSSS pressure vessel and components (Section 4.3.1), ASME Section III Class 1 piping and components (Section 4.3.2) and reactor vessel internals (Section 4.3.3). Without these values, the staff cannot ascertain

whether the CUF for any location exceeded the allowable limit or evaluate the applicant's dispositions of these TLAA in accordance with 10 CFR 54.21(c).

The staff noted that FSAR Table 3.9(B)-21 provides a CUF value of 0.95 for the ASME Section III, Class 1 RCS pressurizer safety and relief valve system. LRA Table 3.1.2-1 states that for "Valve Body (Class 1)", TLAA is used to manage the aging effect of cumulative fatigue damage. However, LRA Section 4.3 did not provide any details of fatigue analyses for RCS Class 1 valves. It is not clear to the staff how the TLAA of all Class 1 valves were dispositioned in accordance with 10 CFR 54.21(c).

Request

1. Provide the original design basis 40-year CUF values and projected 60-year CUF values for all components and/or critical locations that are applicable to the dispositions in LRA Sections 4.3.1, 4.3.2, and 4.3.3. Justify that TLAA disposition associated with the aging effect of cumulative fatigue damage in accordance with 10 CFR 54.21(c)(1) is appropriate.

2. Clarify and justify the disposition of all Class 1 valves that are TLAAs in accordance with 10 CFR 54.21(c).

3. Clarify whether the fatigue analyses as part of the original design or supplemental fatigue analyses included Class 1 valves. If RCS Class 1 valves were included, provide the CUF value of record for the valves, the design transients and the number of cycles assumed in the fatigue analyses and justify how the TLAA associated with the aging effect of cumulative fatigue damage of Class 1 valves will be dispositioned in accordance with 10 CFR 54.21(c)(1). If RCS Class 1 valves were not included, clarify why they were excluded and justify how age-related degradation of these RCS Class 1 valves will be managed during the period of extended operation.

RAI 4.3.2-1

Background

LRA Section 4.3.2.2 provides the TLAA fatigue assessment of the Seabrook Station, Unit 1 (Seabrook), Pressurizer Surge Line (PSL), Pressurizer Surge Nozzle (PSN), and Hot Leg Surge Line Nozzle (HLSN) subject to thermal stratification/striping in addition to the original design transients.

The applicant concludes that the fatigue analyses (without environmental effects) remain valid for the period of extended operation for the PSL, PSN, and HLSN, in accordance with 10 CFR 54.21(c)(1)(i).

Issue

LRA Section 4.3.2.2 states that the PSL piping was previously evaluated for the effects of thermal stratification and plant-specific transients in 1990, and it was determined that the PSL will remain within the ASME Code requirements for the design life of the unit. However, no further details of this analysis were provided in the LRA, therefore the staff is not able to verify these statements and the applicability of thermal stratification during the extended period of operation. The staff is not able to confirm the applicant's conclusion and disposition for the TLAAs of these Class 1 RCS components.

LRA Section 4.3.2.2 states that as the evaluation of the structural weld overlay applied to the PSN included an elastic-plastic formulation and resulted in CUF less than 1.0 at the PSN. The staff was not able to verify if the elastic-plastic analyses provided the appropriate reduction of conservatism in prior evaluations of structural weld overlay as approved corrective action.

Request

1. Provide the current CUF values of record, without environmental effects, for all the components and/or critical locations that are applicable to LRA Section 4.3.2.2, including the PSL and the hot leg surge line nozzle safe-end. Provide the corresponding CUF values from the original design basis analyses, and from the updated analyses for a projected 60-year operation.

2. Clarify how the elastic-plastic formulation was used in the evaluation and how the elastic-plastic analyses provided appropriate reduction of conservatism existed in previous evaluations of structural weld overlay.

3. Justify the conclusion that the analyses for the pressurizer surge line, pressurizer surge nozzle and hot leg surge line Nozzle remains valid for the period of extended operation, is appropriate for these components. As part of the justification, provide sufficient details, including the type, severity and number of transients, used 1) in the original analysis for design life; 2) in the modified or updated analysis for thermal stratification/striping events; and 3) for the extended period of operation, for the PSL piping, the hot leg surge line nozzle safe-end and the hot-leg surge nozzle-to-pipe weld.

RAI 4.3.3-1

Background

LRA Section 4.3.3 states that for assuring continued functionality of the reactor vessel internals during the desired operating period, including license renewal, it is essential to demonstrate that the effects of aging are adequately managed. The applicant stated that the EPRI "Materials Reliability Program (MRP) Reactor Internals Inspection and Evaluation Guidelines," MRP-227, is intended to support that demonstration. The LRA further states that the PWR Vessel Internals Program will manage the aging effects including changes in dimensions, cracking, loss of fracture toughness, and loss of preload of the reactor vessel internals components for the period of extended operation per 10 CFR 54.21(c)(1)(iii).

The Standard Review Plan-License Renewal (SRP-LR) Section 4.3.1.1.1 states that ASME Class 1 components, which include core support structures, are analyzed for metal fatigue, and ASME Section III requires a fatigue analysis for Class 1 components that considers all transient loads based on the anticipated number of transients. SRP-LR Section 4.3.2.1.1.3 states that in Chapter X of the Generic Aging Lessons Learned (GALL) Report, the staff has evaluated a program for monitoring and tracking the number of critical thermal and pressure transients for the selected reactor coolant system components. The staff has determined that this program is an acceptable aging management program (AMP) to address metal fatigue of the reactor coolant system components according to 10 CFR 54.21(c)(1)(iii).

Issue

The Fatigue Monitoring Program is the only option recommended in the GALL Report to manage the aging effects of metal fatigue, and any other option such as a comprehensive inspection and flaw tolerance evaluation program based on the EPRI report MRP-227 guidelines needs to be evaluated on a case-by-case basis. The staff noted that the EPRI report MRP-227 has not been approved by the NRC.

To disposition fatigue TLAA for reactor core internals in accordance with 10 CFR 54.21(c)(1)(iii), an acceptable inspection/flaw tolerance evaluation program should at least include the following elements:

1. Fatigue CUF analysis of all fatigue sensitive reactor internals components to identify the fatigue limiting locations. The fatigue sensitive locations should include upper and lower core plates, baffle-former bolts, core barrel outlet nozzle weld, thermal shield flexures, control rod guide tube (lower flange and support pins), upper support ring or skirt, and any other plant-specific fatigue sensitive component.

2. Comprehensive inspection program that ensures (a) all the fatigue sensitive locations identified in the CUF analyses can be inspected and (b) the inspection techniques are adequate to detect fatigue cracks at all of those locations.

3. Postulated flaw tolerance evaluation to establish acceptable inspection interval. Since neutron irradiation has a significant effect on fracture toughness and crack growth rates of austenitic stainless steels in pressurized water reactor (PWR) environments, an acceptable flaw tolerance evaluation should include: (a) end-of-life neutron fluence for all locations identified in item 1 above, and estimates of reduction in fracture toughness corresponding to those fluence levels, and (b) acceptable crack growth rates for irradiated materials in PWR coolant environments under both corrosion fatigue and stress corrosion cracking conditions.

It is not clear that Seabrook's proposed inspection program can ensure detection of cracks at fatigue sensitive locations, and whether all fatigue locations in the vessel internals components can be inspected.

Request

1. If the fatigue TLAAs for reactor core internals are to be dispositioned in accordance with the criterion of 10 CFR 54.21(c)(1)(iii) using proposed AMP other than the GALL AMP X.M1, provide a comprehensive inspection and postulated flaw tolerance evaluation program that includes at least the three elements discussed above. Justify that the proposed AMP can adequately manage the aging effect of metal fatigue for the reactor vessel internals during the period of extended operation.

2. If the fatigue TLAAs for reactor core internals are to be dispositioned using the Metal Fatigue Of Reactor Coolant Pressure Boundary in accordance with 10 CFR 54.21(c)(1)(iii), or in accordance with either 10 CFR 54.21(c)(1)(i) or

10 CFR 54.21(c)(1)(ii), provide the updated CUFs for the period of extended operation for the reactor internals components for which there is a fatigue CUF of record and justify the proposed disposition.

RAI 4.3.3-2

Background
In LRA Table 3.1.2-3, the applicant indicated that a TLAA disposition is used for the flux thimbles tube and flux thimble guide tubes for the aging effect of cumulative fatigue damage.

Issue
LRA Section 4.3.3, did not provide the CUF values for the flux thimble tubes and flux thimble guide tubes to support the disposition of the TLAA in accordance with 10 CFR 54.21(c)(1)(iii).

Request
Provide the CUF values for the flux thimble tubes and flux thimble guide tubes and justify that the aging effect of cumulative fatigue damage of the flux thimble tubes and flux thimble guide tubes will be adequately managed by the PWR Vessel Internal Program in accordance with 10 CFR 54.21(c)(1)(iii).

RAI 4.3.4-1

Background
In LRA Section 4.3.4, the applicant discussed the methodology to determine the locations that require environmentally assisted fatigue analyses consistent with NUREG/CR-6260 "Application of NUREG/CR-5999 Interim Fatigue Curves to Selected Nuclear Power Plant Components." The staff recognized that, in LRA Table 4.3.4-1, there are seven plant-specific locations listed based on the six generic components identified in NUREG/CR-6260. Footnote 2 of Table 4.3.4-1 indicated that the plant-specific locations listed are the limiting location within the boundary of the applicable NUREG/CR-6260 component.

Issue
The GALL Report AMP X.M1, "Metal Fatigue of Reactor Coolant Pressure Boundary" states that the impact of the reactor coolant environment on a sample of critical components should include the locations identified in NUREG/CR-6260 as a minimum, and that additional locations may be needed. During its review, the staff recognized that in footnote 2 of Table 4.3.4-1, the applicant stated that these locations are plant-specific limiting locations within the boundary of the applicable NUREG/CR-6260 component. However, CUF values for other locations were not available in the LRA and the applicant did not provide any justification that these are the limiting locations. Furthermore, the staff noted that the applicant's plant-specific configuration may contain locations that should be analyzed for the effects of the reactor coolant environment other than those identified in NUREG/CR-6260. This may include locations that are limiting or bounding for a particular plant-specific configuration, or that have calculated CUF values that are greater when compared to the locations identified in NUREG/CR-6260.

The staff noted that LRA Section 4.3.2.2 stated that the controlling fatigue location was the hot leg surge line nozzle safe-end. However, footnote 2 in LRA Table 4.3.4-1 indicates the hot-leg surge nozzle-to-pipe weld to be the plant-specific limiting location within the boundary of the applicable NUREG/CR-6260 component location.

The staff also noted that for some RCS locations, the projected 60-y transient cycles yield CUFs that are higher than the design limit of 1.0. Consequently, the transient cycle limit when the CUF for a specific location exceeds 1.0 would vary for the high fatigue usage locations.

Request

1. Provide the 40-year design CUF value and the projected 60-year CUF values (with and without environmental effect) for the hot leg surge line nozzle safe-end. Jusitfy that the hot-leg surge nozzle-to-pipe weld is the limiting location for the pressurizer surge line.

2. Justify that the plant-specific locations listed in LRA Table 4.3.4-1 are bounding for the generic NUREG/CR-6260 components.

3. Confirm and justify that the locations selected for environmentally assisted fatigue analyses in LRA Table 4.3.4-1 consist of the most limiting locations for the plant (beyond the generic components identified in the NUREG/CR-6260 guidance). If these locations are not bounding, clarify the locations that require an environmentally assisted fatigue analysis and the actions that will be taken for these additional locations. If the limiting location identified consists of nickel alloy, state whether the methodology used to perform the environmentally-assisted fatigue calculation for nickel alloy is consistent with NUREG/CR-6909. If not, justify the method chosen.

RAI 4.3.4-2

Background

LRA Section 4.3.4 states that based on NUREG/CR-6260, plant-specific components were identified and the design ASME fatigue usage factors were adjusted by the environmentally-assisted fatigue factors (F_{en}) to obtain the environmentally-assisted fatigue CUFs for the reactor vessel inlet and outlet nozzles, reactor vessel shell and lower head and RHR hot leg nozzle. The results for a 60-year operation are summarized in LRA Table 4.3.4-1. LRA Section 4.3.4 states that for the reactor vessel shell and lower head and reactor vessel inlet and outlet nozzles the CUFs were determined using the number and severity of the design cycles of record, which bound the projected 60-year cycles. For these components, the environmental fatigue effects were determined using the maximum F_{en}, i.e., the slowest strain rate. The temperature is >200°C and dissolved oxygen is <50 ppb. The applicant further stated that for the remainder NUREG/CR-6260 locations, i.e., pressurizer surge line nozzle, charging nozzle, safety injection nozzle, and residual heat removal (RHR) system Class 1 piping, the CUFs were determined using the projected 60-year cycles and environmental effects were incorporated using an effective F_{en}, which was determined with a temperature of >200°C, dissolved oxygen <50 ppb, and the strain rate calculated by the integrated strain rate method from EPRI Report MRP-47, Rev. 1. LRA Section 4.3.4 states that the fatigue analyses indicate that the 60-year CUF in air for all locations is less than the ASME design limit of 1.0. However, the 60-year CUF adjusted for environmental effects is greater than 1.0 for the hot-leg surge line nozzle and the charging nozzle.

Issue
LRA Subsection 4.3.4 does not provide sufficient details on how the CUFs in air and the associated values of environmental fatigue F_{en}, were determined for the NUREG/CR-6260 locations listed in Table 4.3.4-1. It is not clear whether all design transients were lumped into the worst-case transient or other transients were also considered in the fatigue analyses.

In addition, it is not clear to the staff whether the CUF analysis for the RHR system Class 1 piping was performed using plant-specific 60-year projected cycles or using the design number and severity cycles. The "Disposition" subsection of LRA Section 4.3.4 states that the RHR system Class 1 piping analysis was based on Seabrook Station specific conditions, whereas footnote 4 of Table 4.3.4-1 states that this analysis was performed using a design number and design-severity cycles.

Request
1. For each location listed in LRA Table 4.3.4-1, clarify whether the 60-year ASME Air-Curve CUF were calculated using ASME Code Section III NB-3200 or NB-3600.

2. For fatigue calculation using the integrated strain-rate method, describe how the transient definitions (i.e., stress vs. time curves) are selected. Justify the use of the integrated strain-rate method in the fatigue calculation and that it provides conservative results. Clarify whether some transients, all transients, or the worst-case transient (with some of the transients being lumped into) were used.

3. Clarify what the Seabrook Station specific conditions as discussed in the "Disposition" subsection of LRA Section 4.3.4 are. Clarify whether the CUF analysis for the RHR system Class 1 piping was performed using Seabrook Station specific conditions, or using the design number and severity cycles as noted in footnote 4 of Table 4.3.4-1.

RAI 4.3.5-1

Background
LRA Section 4.3.5 summarizes the TLAA for steam generator tubing associated with loss of material due to wear at supports and fatigue in the U-bend region, which is the result from flow-induced vibrations. The applicant's disposition for these TLAAs is in accordance with 10 CFR 54.21(c)(1)(i), that the analyses remain valid for the period of extended operation.

Issue
The applicant discussed that the maximum 40-year wear is less than 0.0050 inch and the maximum 60-year tube wear will be 0.0075 inch. It appears the applicant applied a factor of 1.5 to the maximum 40-year wear to determine the 60-year wear projection. Furthermore, it appears the TLAA evaluations of tube wall wear are based on results of a 40-year period and linearly extrapolated to 60-year extended period of operation. In this case, the applicant has made at least two assumptions which are: 1) wear and fatigue damage progress linearly with time and 2) the key causative factors or operating conditions, such as the flow pattern and the

support gaps, remain unchanged or are bounded by the analyses. However, over time, due to the wear itself and due to the build-up of corrosion products, or due to intentional operational changes, the flow patterns are likely to change. The applicant did not address or include these considerations in making the extrapolation from a 40-year analysis.

In addition, LRA Section states "Low-cycle fatigue usage for the most limiting tube in the most limiting power-uprated operating condition resulting from the flow-induced vibration tube bending stress is 0.2 ksi." This statement is unclear. It appears to mean that the low-cycle fatigue usage is 0.2 ksi. Fatigue usage, however, is a dimensionless value.

<u>Request</u>

1. Clarify the methodology that was used to determine the 60-year projection for the tube wear. Provide the technical basis and justify the linear extrapolation and analysis envelop for likely changes in causative factors for the wear and fatigue that are caused by the flow-induced vibrations or fluid-structure interactions during the extended period of operation.

2. Indicate if any independent confirmation has occurred, or if any plan will occur in the future to ensure the validity of the analyses and its adequacy for the period of extended operation. Alternatively, justify the adequacy of these results from this analysis, for the period of extended operation.

3. Considering that fatigue usage is a dimensionless value, clarify the value for the low-cycle fatigue usage and the induced bending stress that is being referenced in LRA Section 4.3.5. Amend the LRA, as applicable, to address this clarification.

<u>RAI 4.3.5-2</u>

<u>Background</u>

LRA Section 4.3.5 summarizes the TLAA for steam generator tubing associated with loss of material due to wear at supports and fatigue in the U-bend region, which is the result from flow-induced vibrations (FIV). The applicant's disposition for these TLAAs is in accordance with 10 CFR 54.21(c)(1)(i), that the analyses remain valid for the period of extended operation.

<u>Issue</u>

LRA Section 4.3 discusses "Metal Fatigue Analysis of Piping and Components", whereas LRA Section 4.3.5 includes loss of material due to wear of tubes from FIV, as a distinct aging effect that is separate from metal fatigue. The staff noted that the first two paragraphs of the "Analysis" subsection relate exclusively to wear. The applicant discussed that the maximum 40-year wear is less than 0.0050 inch and the maximum 60-year tube wear will be 0.0075 inch. The applicant then concluded that the estimated maximum 60-year tube wall wear will be less than the acceptance criteria of 40% of wall thickness. The staff noted that the applicant did not provide the wall thickness of the tube to justify the statements in LRA Section 4.3.5.

<u>Request</u>

1. Provide the tube wall thickness and demonstrate that the estimated maximum 60-year tube wall wear is under the limit of acceptability of 40% of wall thickness.

2. Justify why the TLAA analysis and disposition associated with tube wall wear should not be included as a stand-alone subsection under LRA Section 4.7 "Plant-specific TLAA," and include it under LRA Table 4.1-1.

RAI 4.3.6-1

Background
LRA Section 4.3.6 states that Alloy 82/182 welds are used for attaching the pressurizer surge, spray, and relief valve nozzles to safe ends, and the safe ends to the connecting piping. The staff noted that for these dissimilar metal welds, complete Alloy 690 structural weld overlays were completed during refueling outage 12 (Spring 2008). The applicant stated that these overlays were supported by fatigue crack growth analyses projected for a 60-year life, to the end of the period of extended operation, and are therefore not TLAAs. LRA Section 4.3.6 also states that a reactor vessel hot-leg nozzle Alloy 600 weld was mitigated through Mechanical Stress Improvement Process (MSIP) repair during refueling outage 13 (Fall 2009). Furthermore, the MSIP repair was also supported by fatigue crack growth analysis projected for a 60-year life, to the end of the period of extended operation, and is therefore not a TLAA.

Issue
The applicant stated that the fatigue crack growth analyses for both the pressurizer and the reactor vessel were projected for a 60 year life to the end of the period of extended operation.

However, the applicant stated in the "Conclusion" portion of LRA Section 4.3.6 that no TLAA exists for fatigue crack growth, fracture mechanics stability, or corrosion analyses. It is not clear to the staff if the analyses performed utilize (1) a 60-year assumption or (2) a 40-year assumption and then projected for a 60-year period.

The applicant also did not demonstrate why such analyses did not meet all of the six criteria identified in 10 CFR 54.3(a) for the definition of a TLAA.

If the analyses were projected for 60 years and it is not clear why the analyses should not be disposed in accordance with 10 CF54.21(c)(1) as a TLAA.

In addition, the LRA does not provide any details regarding the fatigue crack growth analyses.

It is not clear to the staff if the crack growth analyses included crack growth due to both corrosion fatigue and stress corrosion cracking (SCC).

Request
1. Clarify whether the fatigue crack growth analyses performed for both the pressurizer and reactor vessel in LRA Section 4.3.6 utilize a 60-year assumption or a 40-year assumption and then projected for a period of 60-year.
2. Clarify and justify whether the fatigue crack growth analyses performed for both the pressurizer and reactor vessel in LRA Section 4.3.6 include both stress corrosion cracking and corrosion fatigue crack growth.

3. For the fatigue crack growth analyses discussed in LRA Section 4.3.6 for the pressurizer, justify why those analyses should not be identified as a TLAA. If the analyses should be considered as a TLAA and dispositioned in accordance with 10 CFR 54.21(c)(1)(i), justify the disposition and provide detail information regarding the initial flaw size, loading cycles assumptions, and critical flaw size. If the analyses should be considered as a TLAA and disposed of in accordance with 10 CFR 54.21(c)(1)(ii), justify the disposition and provide the information regarding the initial flaw size, loading cycles assumptions, critical flaw size, and projected flaw size through the end of the period of extended operation.

4. For the fatigue crack growth analyses discussed in LRA Section 4.3.6 for the reactor vessel, justify why those analyses should not be identified as a TLAA. If the analyses should be considered as a TLAA and disposed of in accordance with 10 CFR 54.21(c)(1)(i), justify the disposition and provide detailed information regarding the initial flaw size, loading cycles assumptions, and critical flaw size. If the analyses should be considered as a TLAA and disposed of in accordance with 10 CFR 54.21(c)(1)(ii), justify the disposition and provide the information regarding the initial flaw size, loading cycles assumptions, critical flaw size, and projected flaw size through the end of the period of extended operation.

RAI 4.3.7-1

Background
LRA Section 4.3.7 discusses the fatigue-related TLAAs of Non-Class 1 piping and components. The applicant stated that these piping and tubing components can be designed in accordance with ASME Code Section III Class 2 and 3.

Issue
SRP-LR Section 4.3.1.1.2 indicates that for piping designed and analyzed to ANSI B31.1, ANSI B31.1 specifies allowable stress levels based on the number of anticipated thermal cycles. As an example, UFSAR Table 3.2-2 indicates that the piping (downstream of Safety Valves) of the Pressurizer Relief Discharge System is designed to ANSI B31.1. LRA Section 2.3.1.1 also identifies that the Pressurizer Relief Tank, pump, heat exchanger, and connected pipes and valves are within the scope of the License Renewal. However, the applicant did not provide any detail regarding applicable ANSI B31.1 piping in the LRA Section 4.3.7

Request
Clarify whether there are any piping, piping components or piping elements, designed and analyzed in accordance with ANSI B31.1, that are within the scope of the license renewal. Justify that LRA Section 4.3.7 provides adequate disposition of fatigue-related TLAA for all non-Class 1 piping and components (including Class 2, Class 3, and ANSI B31.1)

RAI 4.7.9-1

Background
LRA Section 4.7.9 addresses the fatigue analysis of canopy seal clamp assemblies and states that the design analysis is a TLAA requiring evaluation for the period of extended operation.

Issue
It is not clear to the staff if the original fatigue analysis referred to fatigue crack initiation or fatigue flaw growth. If the analysis was a fatigue flaw growth analysis, detailed information regarding the initial flow size, loading cycles assumption, and critical flow size are needed for the staff to evaluate the TLAA disposition. The staff reviewed LRA Section 3 and Section 4.7.9 and noted that the effect of aging of the canopy seal clamp assemblies was also not identified in the LRA. The staff noted that the fatigue analysis was based on the consideration of 400 cycles consisting of 20 occurrences of the Operating Basis Earthquake (OBE) and each occurrence having 20 cycles of maximum response as discussed in LRA Section 4.7.9. However, in LRA Table 4.3.1-3, the 60-year projected cycles is 10 for OBE and Note 3 of Table 4.3.1-3 indicates that each earthquake has 10 cycles.

Request
1. Clarify the effect of aging for the canopy seal clamp assemblies in LRA Section 4.7.9 and justify that the TLAA disposition is appropriate for the effect of aging of the canopy seal clamp assemblies.

2. Clarify whether the fatigue analysis referred to in LRA Section 4.7.9 is a fatigue crack initiation analysis or a fatigue flaw growth analysis. If the analysis involved is a fatigue flaw growth, justify the disposition and provide detailed information regarding the initial flaw size, loading cycles assumptions, and critical flaw size and justify that the analysis and the disposition of the TLAA is appropriate for the effect of aging of the canopy seal clamp assemblies.

3. Clarify and justify the use of different assumed number of cycles during an OBE earthquake in different sections of the LRA. Clarify whether OBE is the only transient input to the fatigue analysis.

RAI 4.7.10-1

Background
LRA Section 4.7.10 summarizes the TLAA for hydrogen analyzer radiation dose analysis. UFSAR Table 6.2-84 defines the Hydrogen Analyzer design parameters and maximum radiation dose limits of $5x10^6$ rads for 40-years of normal operation.

Issue
The staff reviewed LRA Section 2.3.2.1 and LRA Table 3.2.2-1, Combustible Gas Control System, and was not able to identify the aging effect for the hydrogen analyzer. The staff was not able to identify any aging management review (AMR) line items specifically associated with the hydrogen analyzers in LRA Table 3.2.2-1.

Request
Identify the aging effect associated with the hydrogen analyzers of the Combustible Gas Control System and justify that the analysis and the disposition of the TLAA is appropriate for this aging effect associated with the hydrogen analyzers. Justify that the AMR results provided in Table 3.2.2-1 adequately address the aging effect for the hydrogen analyzers.

RAI 4.7.11-1

Background
LRA Section 4.7.11 states that the evaluation demonstrated that safety-related active mechanical equipment in harsh environments has been adequately addressed. The applicant also stated that mechanical equipment qualification (MEQ) is a TLAA because a period of 40 years was used for normal service radiation exposure. The applicant stated that the temperature, pressure, and time profiles have been adjusted to account for approved power uprate condition. Therefore, no further TLAA evaluations for license renewal are required. However, the staff noted that in UFSAR Section 3.11.2, the environmental parameters of interest are temperature, pressure, humidity, radiation, chemical spray, and submergence. The applicant stated that the effect of aging on the intended function of equipment will be adequately addressed for the period of extended operation. Commitment No. 45 was provided in the LRA indicating that the MEQ files will be revised prior to the period of extended operation.

Issue
SRP-LR Section 4.7.3.1.2 indicates that for a TLAA disposition pursuant to 10 CFR 54.21(c)(1)(ii), the applicant shall provide a sufficient description of the analysis and document the results of the reanalysis to show that it is satisfactory for the 60-year period. The application does not provide this information. Therefore, it is not clear to the staff if the applicant accounted for the environmental parameters in addition to radiation exposure, temperature, pressure, and time profiles. It is also not clear to the staff what the "time profiles" are. For the radiation exposure, the applicant also did not identify the 40-year radiation exposure limit of the safety-related active mechanical equipment, the projected 60-year radiation exposure limit, or the design limit for the radiation exposure. Without such information, the staff cannot evaluate the adequacy of the TLAA.

Furthermore, it is not clear to the staff how the applicant's Commitment 45 is consistent with disposition of the TLAA pursuant to 10 CFR 54.21(c)(1)(ii). There is also no indication of what information will be re-evaluated and revised in Commitment No. 45.

Request
1. Clarify and justify that all environmental parameters identified in UFSAR 3.11.2 has been evaluated and accounted for a period of 60 years. Clarify in detail what the time profile is being referred to in LRA Section 4.7.11.

2. Identify the 40-year radiation exposure limit of the safety-related active mechanical equipment, the projected 60-year radiation exposure limit, and the design limit for the radiation exposure. Justify that the radiation exposure have been appropriately accounted for in the TLAA analysis.

3. 10 CFR 54.21(c)(1)(ii) requires a demonstration that analyses have been projected to the end of the PEO. Explain how LRA Section 4.7.11 and Commitment 45 satisfy 10 CFR 54.21(c)(1)(ii).

4. Clarify and revise Commitment No. 45 to delineate the information to be re-evaluated and revised. Clarify whether all the MEQ files will be revised or justify why only selected MEQ files will be revised.

RAI 4.1-1

Background
LRA Table 4.1-2 "Review of Analyses Listed in NUREG-1800 Table 4.1-3 – Additional Examples of Plant-Specific TLAAs," states that the analysis associated with flow-induced vibration (FIV) endurance limit is applicable and is addressed in LRA Section 4.3.3.

UFSAR Section 3.9(N).2.4 discusses the impact of FIV effects on the integrity of the reactor vessel internal (RVI) components.

Issue
LRA Section 3.1 and LRA Table 3.1.2-1 does not indicate whether FIVs can result in either cracking by high cycle fatigue or loss of material by fretting or wear of the RVI components that are identified in LRA Table 3.1.2-2. In addition, LRA Section 4.3.3 does not discuss how cracking induced by FIVs and loss of material induced by FIVs is being managed in the RVI components that are in the scope of license renewal and listed in LRA Table 3.1.2-2.

It is not clear to the staff whether cracking induced by FIVs and loss of material induced by FIVs are applicable aging effects requiring management (AERM) for the RVI components in LRA Table 3.1.2-3 and if so, whether the CLB includes any analysis that evaluated these aging effects and mechanisms on the integrity of the RVI components. It is also not clear whether the analysis is a TLAA as defined in 10 CFR 54.3.

Request
Clarify whether cracking induced by FIVs or loss of material (i.e., wear or fretting) induced by FIVs are applicable AERM for the RVI components that are subject to AMR in LRA Table 3.1.2-2.

Clarify whether the CLB includes any analysis that evaluated the impact of FIVs on the structural integrity of the RVI components as a result of these aging mechanisms. If so, clarify and justify whether the analysis is a TLAA as defined in 10 CFR 54.3 and whether the LRA needs to be amended to include this analysis in accordance with 10 CFR 54.21(c)(1).

RAI 4.1-2

Background
LRA Table 4.1-2 "Review of Analyses Listed in NUREG-1800 Table 4.1-3 – Additional Examples of Plant-Specific TLAAs," states that the analysis associated with ductility reduction of

fracture toughness for the RVI components is applicable and addressed in LRA Section 4.3.3. However, LRA Section 4.3.3 does not include a discussion related to this specific issue.

UFSAR Section 3.9(N).5.4 indicates that the RVI core support structure components have been designed to ASME Section III, Subsection NG. ASME Section III Paragraph NG-2160 states that it is the responsibility of the owner to select material suitable for the conditions stated in the Design Specification (NA-3250), with specific attention being given to the effects of service conditions upon the properties of the material.

LRA Table 3.1.2-2 identifies loss of fracture toughness as an applicable aging effect requiring management AERM for the RVI components that are subject to an AMR. The applicant's further evaluation discussion in LRA Section 3.1.2.2.6 is associated with these AMR items.

Issue
The service conditions that may degrade material properties of RVI components include neutron irradiation, temperature, and reactor coolant environment. Recent industry reports, such as MRP-211, "Materials Reliability Program: PWR Internals Age-Related Material Properties, Degradation Mechanisms, Models, and Basis Data—State of Knowledge - TR-1015013, show that fracture toughness of irradiated stainless steels is decreased to a K_{1c} = 38 MPa $m^{1/2}$ at neutron fluencies of 5 dpa. The staff noted that most RVI components are likely to exceed this neutron dose level during the period of extended operation. It is not clear to the staff whether the CLB includes any neutron fluence-dependent reduction of fracture toughness analysis for the RVI components that are in the scope of license renewal and listed in LRA Table 3.1.2-2.

Request
Clarify whether the CLB includes any neutron fluence-dependent reduction of fracture toughness analysis for the RVI components that are subject to an AMR in LRA Table 3.1.2-3. If so, clarify and justify whether the analysis is a TLAA as defined in 10 CFR 54.3 and whether the LRA needs to be amended to include this analysis in accordance with 10 CFR 54.21(c)(1).

RAI 4.1-3

Background
LRA Section 4.1.3 states that pursuant to 10 CFR 54.21(c)(2), an applicant for license renewal should include a list of unit-specific exemptions granted to 10 CFR 50.12 that are in effect and based on a TLAA as defined in 10 CFR 54.3. The applicant stated that each exemption has been reviewed to determine exemptions that are based on a TLAA. The applicant further stated that the CLB documentation, identified in Section 4.1.1, was reviewed and no exemptions granted pursuant to 10 CFR 50.12 and based on a TLAA as defined in 10 CFR 54.3, have been identified.

On November 12, 2002, FPL Energy Seabrook, LLC, requested an exemption from compliance with the pressure-temperature (P-T) limit generation requirements of 10 CFR Part 50, Appendix G. The applicant requested this exemption in accordance with 10 CFR 50.60(b), specifically, requesting NRC approval to use ASME Code Case N-641 as the basis for generating its P-T limit curves. The staff granted this exemption in accordance with

10 CFR 50.12 by its safety evaluation and exemption approval letter to FPL Energy Seabrook, LLC, dated August 1, 2003.

The staff noted that the exemption permits the applicant to generate the P-T limit curves using the adjusted reference temperature equation for a K_{1c} linear elastic fracture toughness criterion.

Issue
The staff noted that the stated exemption has a specific relationship to the applicant's generation of its P-T limit curves, which is identified as a TLAA and documented in LRA Section 4.2.4. The staff's review did not identify a withdrawal of the exemption that was granted in accordance with 10 CFR 50.12 by its safety evaluation and exemption approval letter to FPL Energy Seabrook, LLC, dated August 1, 2003. Therefore, the staff noted that this exemption may need to be identified as an exemption that is based on a TLAA in accordance with 10 CFR 54.21(c)(2).

Request
Clarify and justify whether the stated exemption needs to be identified as an exemption that is based on a TLAA in accordance with 10 CFR 54.21(c)(2) and whether continuation of the exemption(s) will be needed for the period of extended operation.

RAI 4.7.12-1

Background
SRP-LR Section 4.1.2 states that pursuant to 10 CFR 54.3, TLAAs are those licensee calculations and analyses that:

1. *Involve systems, structures, and components within the scope of license renewal, as delineated in 10 CFR 54.4(a);*

2. *Consider the effects of aging;*

3. *Involve time-limited assumptions defined by the current operating term, for example, 40 years;*

4. *Were determined to be relevant by the licensee in making a safety determination;*

5. *Involve conclusions or provide the basis for conclusions related to the capability of the system, structure, or component to perform its intended function(s), as delineated in 10 CFR 54.4(b); and*

6. *Are contained or incorporated by reference in the CLB.*

LRA Section 4.7.12 summarizes the absence of TLAAs for metal corrosion allowances and corrosion effects. The applicant stated that a review of the Seabrook Station licensing basis found no description of time dependent corrosion allowances, rates, or corrosion-dependent design lives of pressure vessels, system components, piping, or metal containment components, and therefore concluded that there are no TLAAs for metal corrosion allowances and corrosion effects.

The Seabrook UFSAR, Section 5.4.2.3, Steam Generators, Design Evaluation, subsection (d), Allowable Tube Wall Thinning under all Plant Conditions, states:

> The corrosion rate is based on a conservative weight loss rate of Inconel tubing in flowing 650°F primary side reactor coolant fluid. The weight loss, when equated to a thinning rate and projected over a 40-year design operating objective, with appropriate reduction after initial hours, is equivalent to 0.083 mils thinning. The assumed corrosion rate of 3 mils leaves a conservative 2.917 mils for general corrosion thinning on the secondary side.

Issue
The staff's review of the Seabrook licensing basis in the UFSAR identified a description of time dependent corrosion associated with the steam generator tubes. The staff believes that the weight loss rate and the remaining wall calculations for a 40-year design meet the six criteria in the SRP-LR and therefore, meet the definition of TLAA.

Request
Provide additional justification as to why metal corrosion allowance for steam generator tube wall is not considered a TLAA.

RAI 4.2.2-1

Background
The Seabrook LRA Tables 4.2.2-1 and 4.2.3-1 include data for the extended beltline materials (including the upper, intermediate, and lower shells of the RPV and the associated welds). The NRC's Reactor Vessel Integrity Database (RVID) does not contain information for the upper shell, the upper shell axial welds, and the upper-to-intermediate shell circumferential weld of the Seabrook RPV.

Request
Discuss the procedures that you used to determine the chemistry data, initial reference temperature (RT_{NDT}), margins and initial upper shelf energy (USE) values for the extended beltline materials to demonstrate that you have applied consistent approaches in determining the above mentioned material information for all of the extended beltline materials.

RAI 4.7.2-1

Background
The applicant is relying on the fatigue crack growth analysis in WCAP-14535-A, "Topical Report on Reactor Coolant Pump Flywheel Inspection Elimination," as the TLAA for the reactor coolant pump (RCP) flywheels. The staff verified that the NRC endorsed the methodology and results in this WCAP report for use in a safety evaluation (SE) dated September 12, 1996. However, in the conclusion section of the SE (Section 4.0), the staff concluded that the inspections of the

flywheels should be performed even if all of the recommendations of Regulatory Guide 1.14, Revision 1, "Reactor Coolant Pump Flywheel Integrity," were met and that the inspections of the RCP flywheels should not be completely eliminated.

Issue
The applicant has not clearly linked the operating experience at Seabrook with the fatigue crack growth analysis in WCAP-14535-A. Plus, it is not clear from the TLAA discussion whether the applicant intends to be consistent with the position taken in the staff's SE of September 12, 1996 and continue the inservice inspection (ISI) of the RCP flywheels during the period of extended operation, or whether the applicant is proposing to discontinue the ISI of the RCP flywheels during the period of extended operation.

Request
1. Discuss the past examination results for the RCP flywheels at Seabrook and how those results justify the use of the WCAP-14535-A.

2. Clarify whether the applicant intends to continue the ISI of the RCP flywheels consistent with the NRC's SE on WCAP-14535, dated September 12, 1996. If ISI will be performed during the period of extended operation, the staff also requests the applicant to justify what type of inspections will be performed on the RCP flywheels during the period of extended operation and the frequency that will be used for the inspections. Otherwise, the applicant is requested to justify its basis for discontinuing the ISI of the RCP flywheels if the ISI will be discontinued during the period of extended operation.

RAI 4.3.1-1

Background
In LRA Section 4.3.1, the applicant discussed the 60-year transient projection methodology. The applicant stated that the 60-year projection was determined by adding the cumulative number of occurrences as of April 1, 2009 to the number of cycles predicted to occur in the 41 years of future operation.

Issue
The applicant provided a summary of the projected number of cycles in LRA Table 4.3.1-3. The staff noted that the "Unit Loading Between 0% and 15% Power" and "Unit Unloading Between 0% and 15% Power" transient cycles listed in that table for the "60-Year Projected Cycles" are not consistent with the current count for these transients. More specifically, the projected numbers (listed as 13 and 10) are smaller than the actual counts so far (listed as 27 and 26) for the "Unit Loading Between 0% and 15% Power" and "Unit Loading Between 0% and 15% Power" transients, respectively.

Request
For the transients "Unit Loading Between 0% and 15% Power" and "Unit Unloading Between 0% and 15% Power" in LRA Table 4.3.1-3, justify why the values of 60-Year Projected Cycles are smaller than the values of "Current Cycles." Clarify the values in the "Current Cycles" column and "60-Year Projected Cycles column." Amend the LRA, as applicable, to address this clarification.

January 5, 2011

Mr. Paul Freeman
Site Vice President
c/o Mr. Michael O'Keefe
NextEra Energy Seabrook, LLC
P.O. Box 300
Seabrook, NH 03874

SUBJECT: REQUEST FOR ADDITIONAL INFORMATION RELATED TO THE REVIEW OF THE SEABROOK STATION LICENSE RENEWAL APPLICATION (TAC NO. ME4028)

Dear Mr. Freeman:

By letter dated May 25, 2010, NextEra Energy Seabrook, LLC, submitted an application pursuant to Title 10 of the *Code of Federal Regulations* Part 54, to renew Operating License NPF-86 for Seabrook Station, Unit 1, for review by the U.S. Nuclear Regulatory Commission (NRC or the staff). The staff is reviewing the information contained in the license renewal application and has identified, in the enclosure, areas where additional information is needed to complete the review.

The request for additional information was discussed with Mr. Rick Cliche, and a mutually agreeable date for the response is within 30 days from the date of this letter. If you have any questions, please contact me at 301-415-1427 or by e-mail at richard.plasse@nrc.gov.

Sincerely,
/RA/

Richard Plasse, Project Manager
Projects Branch 2
Division of License Renewal
Office of Nuclear Reactor Regulation

Docket No. 50-443

Enclosure:
As stated

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Letter to P. Freeman from R. Plasse dated January 5, 2011

SUBJECT: REQUEST FOR ADDITIONAL INFORMATION RELATED TO THE REVIEW OF THE SEABROOK STATION LICENSE RENEWAL APPLICATION (TAC NO. ME4028)

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