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Submitted: August 20, 2012

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

December 14, 2010

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) – AGING MANAGEMENT PROGRAMS

Dear Mr. Freeman:

By letter dated May 25, 2010, NextEra Energy Seabrook, LLC (NextEra) submitted an application pursuant to Title 10 of the *Code of Federal Regulations* Part 54, to renew the Operating License NPF-86 for Seabrook Station, Unit 1 (Seabrook) 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 4
Aging Management Programs

RAI B.2.1.1-1

Background
The applicant's ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD Program states that the aging management program (AMP) is "an existing program consistent with NUREG-1801, Section XI.M1." GALL AMP XI.M1 recommends the use of American Society of Mechanical Engineers (ASME) Section XI Table IWB-2500-1 to determine the examination of Category B-F and B-J welds. The applicant is currently including applicable portions of the Categories B-F and B-J in its Risk Informed Inservice Inspection Program.

Issue
The staff noted that the approval of the risk-informed methodology cannot be assumed for subsequent ten-year intervals.

Request
Clarify how the inspection of Categories B-F and B-J will be implemented as part of the ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD Program during the period of extended operation.

RAI B.2.1.3-1

Background
The "preventive actions" program element of GALL AMP XI.M3, "Reactor Head Closure Studs," references the guidance outlined in RG 1.65 originally issued in 1973. RG 1.65, Rev. 1 was issued in April 2010 and includes using bolting material for closure studs that has a measured yield strength less than 150 ksi, which is resistant to stress corrosion cracking.

LRA Section B.2.1.3 states that the Seabrook reactor head closure studs are manufactured from SA-540, Class 3, Grade B24 material and the maximum tensile strength of the material is less than 170 ksi as recommended in GALL Report, Rev. 1.

Issue
LRA Section B.2.1.3 does not include the preventive action of using stud materials with a measured yield strength level less than 150 ksi in comparison with RG 1.65, Rev. 1. The staff needs to confirm whether the applicant's program considers the strength levels of reactor head closure stud materials as addressed in the RG 1.65, Rev. 1 to adequately manage stress corrosion cracking.

Request

1) Clarify whether the measured yield strength of the reactor head closure stud material used at Seabrook Station exceeds 150 ksi.

2) Are there program provisions that would preclude use of materials with yield strength greater than 150 ksi? If not, or if the reactor head closure stud material has a yield strength level greater than or equal to 150 ksi, justify the adequacy of the Reactor Head Closure Studs Program to manage stress corrosion cracking in the high-strength material.

RAI B.2.1.3-2

Background
The program description of GALL AMP XI.M3, "Reactor Head Closure Studs," states that the recommended program includes inservice inspection to detect cracking, loss of material and coolant leakage from reactor head closure studs. The "preventive actions" program element of GALL AMP XI.M3 also includes using manganese phosphate or other acceptable surface treatments and stable lubricants. LRA Section B.2.1.3 indicates that a station approved lubricant is utilized during installation/removal of the studs that does not contain molybdenum disulfide (MoS_2).

Issue
Operating Experience No. 2 described in LRA Section B.2.1.3 states that discoloration was reported on some of the reactor head closure studs during Refueling Outage 8 in 2002, and that the discoloration was due to the lubricant used for stud removal and was considered not an indication of stud degradation. During the staff's audit, the applicant also stated that the substance applied on the studs was WD-40. The staff needs to confirm whether the discoloration is related to an age-related degradation. The staff also needs to clarify whether WD-40 is a stable lubricant at the operating temperatures and compatible with reactor bolting materials and environment.

Request

1) Clarify what the root cause for the discoloration on the studs was and whether the discoloration has been repeatedly observed. In your response, provide further justification why the observed discoloration is not associated with an aging effect that requires management during the period of extended operation, such as loss of material due to corrosion or wear. If the discoloration is associated with an aging effect, justify how it will be managed during the period of extended operation.

2) Provide the service temperature range of the lubricant based on its technical specification or equivalent. In addition, compare the service temperature with the operating temperatures of the reactor head closure studs. In view of the foregoing evaluation, further clarify whether the lubricant is stable at the operating temperatures and is compatible with the stud and vessel materials and with the surrounding environment.

RAI B.2.1.7-1

Background
SRP-LR, Section 3.1.2.2.17, "Cracking due to Stress Corrosion Cracking, Primary Water Stress Corrosion Cracking, and Irradiation-Assisted Stress Corrosion Cracking," states that the applicant should provide a commitment in the FSAR Supplement to:

1. Participate in the industry programs for investigating and managing aging effects on reactor internals;

2. Evaluate and implement the results of the industry programs as applicable to the reactor internals; and

3. Upon completion of these programs, but not less than 24 months before entering the period of extended operation, submit an inspection plan for reactor internals to the NRC for review and approval.

Issue
LRA Section B.2.1.7, "PWR Vessel Internals," summarizes the industry and plant-specific operating experience in the "operating experience" program element. However, the applicant does not mention any issues associated with cracking of components fabricated from Alloy X-750.

Request
Identify the components in the reactor vessel internals that are fabricated from Alloy X-750. Discuss the plant-specific experience associated with PWR vessel internal components fabricated from Alloy X-750. Furthermore, discuss the future plans for managing age-related degradation in those components fabricated from Alloy X-750.

RAI B.2.1.8-1

Background
The parameters monitored or inspected program element of the GALL AMP XI.M17, "Flow-Accelerated Corrosion," indicates that the effects of flow accelerated corrosion on the intended function of piping and components are monitored by measuring wall thickness. LRA Section B.2.1.8 states that valves, orifices, equipment nozzles, and other like components that cannot be inspected completely with ultrasonic techniques due to their shape and thickness are evaluated based on the wear of piping located immediately downstream.

The program guidance document, NSAC-202L, states that this approach is only applicable if the piping downstream is manufactured of material with equal or higher susceptibility, and has not been repaired or replaced. It also recommends that the piping be inspected for two diameters downstream of the connecting weld, and, if possible, a portion of the component itself. It continues by stating that if significant wear is detected in the downstream pipe, that the component should also be examined, and that a combination of ultrasonic testing, radiography, and/or visual techniques are typically utilized to inspect these components.

Issue
It is not clear whether the program will implement the recommendations in NSAC 202L for components that cannot be completely inspected with ultrasonic testing and how the follow-up inspections will be implemented in the Flow-Accelerated Corrosion Program.

Request
Provide additional information regarding (1) inspection of the valves, orifices, equipment nozzles, and other like components that cannot be inspected completely with ultrasonic techniques due to their shape and thickness if significant wear is detected in piping located immediately downstream and (2) how the follow-up inspections will be implemented in the Flow-Accelerated Corrosion Program.

RAI B.2.1.10-1

Background
SRP-LR Section 3.1.2.2.16 identifies that cracking due to primary water stress corrosion cracking (PWSCC) could occur on the primary coolant side of PWR steam generator (SG) tube-to-tubesheet welds made or clad with nickel alloy. The Generic Aging Lessons Learned (GALL) Report recommends American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code) Section XI Inservice Inspection (ISI) and control of water chemistry to manage this aging. The GALL Report recommends no further aging management review for PWSCC of nickel alloy if the applicant complies with applicable U.S. Nuclear Regulatory Commission (NRC) Orders and provides a commitment in the Final Safety Analysis Report (FSAR) supplement to implement applicable: (1) Bulletins and Generic Letters and (2) staff-accepted industry guidelines. In GALL Report, Revision 1, Volume 2, this aging is addressed in item IV.D2-4 as applicable only to once-through steam generators, but not to recirculating steam generators.

The staff notes that ASME Code Section XI does not require any inspection of the tube-to-tubesheet welds. In addition, no specific NRC Orders, or bulletins require any examination of this weld. The staff's concern is that, if the tubesheet cladding is Alloy 600 (i.e., Alloy 82/182 weld metal), the tube-to-tubesheet weld region may not have sufficient Chromium content to prevent initiation of PWSCC. Consequently, such a crack initiated in this region (i.e., close to a tube) could propagate into the weld, causing a failure of the weld and of the primary-secondary pressure boundary. Thus, this aging effect may potentially impact both once-through and recirculating steam generators.

In LRA Table 3.1.1, the applicant stated that item 3.1.1-35 is not applicable because they do not have once-through steam generators and therefore, do not have the components associated with this model of SGs. In UFSAR Section 5.4.2.4, the applicant stated that the Seabrook Model F SGs contain thermally treated Alloy 600 tubes and that the primary side of the SG tubesheet is weld clad with Inconel (i.e., Alloy 82/182).

Issue
Unless the NRC has approved a redefinition of the pressure boundary in which the autogenous tube-to-tubesheet weld is no longer included, the staff considers that the effectiveness of the primary water chemistry program should be verified to ensure PWSCC cracking is not occurring.

Request

1. For Seabrook Model F SGs, clarify whether the tube-to-tubesheet welds are included in the reactor coolant pressure boundary or alternate repair criteria have been permanently approved.

2. If there is no alternate repair criteria permanently approved, provide a plant-specific AMP, along with the primary water chemistry program or justify on alternative method to manage this potential aging effect and to ensure that cracking due to PWSCC is not occurring in tube-to-tubesheet welds.

RAI B.2.1.10-2

Background
SRP-LR Section 3.1.2.2.13 identifies that cracking due to PWSCC could occur in PWR components made of nickel-alloy and steel with nickel-alloy cladding, including reactor coolant pressure boundary components and penetrations inside the RCS such as pressurizer heater sheathes and sleeves, nozzles, and other internal components. GALL Report, Revision 1, Volume 2, item IV.D1-6 recommends AMP XI.M2, "Water Chemistry," for PWR primary water for managing the aging effect of cracking in the nickel alloy SG divider plate exposed to reactor coolant.

LRA Table 3.1.1, item 3.1.1-81, credits the Water Chemistry Program to manage cracking due to primary stress corrosion cracking in nickel-alloy steam generator primary channel head divider plate exposed to reactor coolant in the steam generators.

Issue
From foreign operating experience in SGs with a similar design to that of Seabrook SGs, cracking due to PWSCC has been identified in SG divider plate assemblies made with Alloy 600, even with proper primary water chemistry. Specifically, cracks have been detected in the stub runner, very close to the tubesheet/stub runner weld and with depths of almost a third of the divider plate thickness. Therefore, the staff notes that the water chemistry program alone does not appear to be effective in managing the aging effect of cracking due to PWSCC in SG divider plate assemblies.

Although these SG divider plate assembly cracks may not have a significant safety impact in and of themselves, such cracks could affect adjacent items that are part of the reactor coolant pressure boundary, such as the tubesheet and the channel head, if they propagate to the boundary with these items. For the tubesheet, PWSCC cracks in the divider plate could propagate to the tubesheet cladding with possible consequences to the integrity of the tube-to-tubesheet welds. For the channel head, the PWSCC cracks in the divider plate could propagate to the SG triple point and potentially affect the pressure boundary of the SG channel head.

Request

1. Please discuss the materials of construction of your SG divider plate assemblies.

   If any constitutive/weld material or base metal of the SG divider plate assemblies is susceptible to cracking (e.g., Alloy 600 or the associated Alloy 600 weld materials), please describe an aging management or inspection program (examination technique and frequency) to ensure that there are no cracks which could propagate into other items which are part of the reactor coolant pressure boundary (e.g., tubesheet and channel head) that could challenge the integrity of those adjacent items.

RAI B.2.1.11-1

Background
The GALL Report AMP XI.M20, "Open Cycle Cooling Water System," program element 1, scope of program, states that the program addresses the aging effects of material loss and fouling due to micro- or macro-organisms and various corrosion mechanisms. The program basis document states that this program will manage hardening and loss of strength due to elastomer degradation. During onsite discussions, the applicant stated that hardening and loss of strength due to elastomer degradation would be identified by visual inspections.

Issue
The detection of hardening or loss of strength due to elastomer degradation appears to be impractical without some type of physical manipulation of the components being managed. It is unclear how the Open Cycle Cooling Water System Program would manage the hardening and loss of strength of elastomer degradation by visual inspections only.

Request
Provide the technical basis on how hardening and loss of strength due to elastomer degradation will be managed by the Open Cycle Cooling Water System.

B.2.1.12-1

Background
The GALL Report AMP XI.M21, "Closed-Cycle Cooling Water System," program element 3, parameters monitored/inspected, states that the aging management program monitors the effects of corrosion and stress corrosion cracking (SCC) by testing and inspection in accordance with guidance in EPRI TR-107396 to evaluate system and component condition. LRA Section B.2.1.12 stated it took an exception to the EPRI guidelines by raising the action level for hydrazine in the thermal barrier system from 200 ppm to 300 ppm. In addition, LRA Section B.2.1.12 stated that it took an exception to the EPRI guidelines by raising the action level for sulfates from 150 ppb to 500 ppb in the thermal barrier system. The discussion in the LRA regarding these exceptions stated that the hydrazine level was increased in 50 ppm increments until 300 ppm was reached "without indication of increased copper corrosion." However, the technical background for increasing the hydrazine level did not appear to discuss

how the copper corrosion was evaluated. For sulfates, the LRA stated that the evaluation of the upper limit of 500 ppb had concluded that the thermal barrier system's low oxygen levels, alkaline pH and absence of sulfides would mitigate the concern regarding the sulfate level above 150 ppb. However, the basis documentation onsite did not provide technical justification for the increased sulfate action level.

Issue
It is not clear from the onsite technical basis documents why it is appropriate to have a higher hydrazine and sulfate action levels in the thermal barrier system than is recommended in the EPRI Closed Cooling Water Chemistry Guideline Report.

Request
Provide justification for the higher hydrazine and sulfate action levels in the thermal barrier system compared to what is recommended in the EPRI Closed Cooling Water Chemistry Guideline Report, and why these higher levels will not lead to enhanced degradation. If sulfate and hydrazine did increase above the action level guidelines in the EPRI Closed Cooling Water Chemistry Guideline Report, provide information on the effect of the excursion on aging during the period of extended operation.

B.2.1.12-2

Background
The GALL Report AMP XI.M21, "Closed-Cycle Cooling Water System," program element 2, preventive actions, states the program maintains system corrosion inhibitor concentrations within the specified limits of the EPRI Closed Cooling Water Chemistry Guideline Report. The EPRI report identifies Action Level 1 and Action Level 2 for the pH level in blended glycol formulations. The report generally describes Action Level 1 as being outside the normal operating level with recommendations to increase the monitoring frequency and to enter Action Level 2 if the parameter has not returned to the normal operating range within 90 days. The Seabrook on-site documentation states that the Diesel Generator Cooling Water Jacket System, (a blended glycol formulation), only has a pH Action Level 2, and does not identify an Action Level 1 for pH.

Issue
It is not clear to the staff why the on-site guidelines for the Diesel Generator Cooling Water Jacket is not consistent with the EPRI Closed Cooling Water Chemistry Guideline Report by having two action levels for pH.

Request
Justify why the pH action levels for the Diesel Generator Cooling Water Jacket is not consistent with that found in the EPRI Closed Cooling Water Chemistry Guideline Report.

B.2.1.12-3

Background
The GALL Report AMP XI.M21, "Closed-Cycle Cooling Water System," program element 3, parameters monitored/inspected, states that the program monitors the effects of corrosion and SCC by testing and inspecting in accordance with guidance in the EPRI Closed Cooling Water Chemistry Guideline Report to evaluate system and component condition. LRA Section B.2.1.12 took an exception to the EPRI guidance recommendation for performance and functional testing to verify the effectiveness of chemistry controls and management of aging effects. According to the LRA, the EPRI guidance notes that performance testing is typically part of an engineering program that verifies a component's active functions, and that these activities would fall under the 10 CFR 50.65, Maintenance Rule. The staff notes that while the EPRI guidance does state that performance monitoring is typically part of an engineering program, it also states that performance monitoring can be used to confirm that the conditions in the closed cooling water system are not degrading heat exchanger performance, and that logging and trending of system parameters is an important part of the closed cooling water system monitoring program.

Issue
It is not clear to the staff if maintenance rule activities are being credited to manage the aging effects in the Closed-Cycle Cooling Water System during the period of extended operation. If so, it is not clear to the staff where the activities contained in the maintenance rule is captured by an aging management program.

Request
Provide information on how the maintenance rule activities are included in the aging management for the Closed-Cycle Cooling Water Systems during the period of extended operation.

B.2.1.12-4

Background
The GALL Report AMP XI.M21, "Closed-Cycle Cooling Water System," program element 6, "acceptance criteria," states that acceptance criteria and tolerances are to be based on system design parameters and functions. Exception No. 5 to the program states that the program does not rely on performance or functional testing to verify the effectiveness. The justification for the exception states that the program uses corrosion monitoring and internal inspections of opportunity to monitor program effectiveness, but also adds that test coupons in several systems are used to check the effectiveness of the corrosion inhibitor by quantifying the corrosion rates of the coupons.

Issue
The onsite basis documents did not contain the acceptance criteria for evaluating the results from the test corrosion coupons and visual inspection surveillance activities.

Request
Provide the acceptance criteria that will be used for the corrosion coupons and visual inspection surveillance activities.

B.2.1.12-5

Background
The GALL Report AMP XI.M21, "Closed-Cycle Cooling Water System" states that the aging management program monitors the effects of corrosion and stress corrosion cracking by testing and inspection in accordance with guidance in EPRI TR-107396, and that the effectiveness of the program is confirmed by visual inspections and performance/functional tests. Exception No. 5 to the program states that the program does not rely on performance or functional testing to verify the effectiveness. The justification for the exception states that the program uses corrosion monitoring and internal inspections of opportunity to monitor program effectiveness, but also adds that test coupons in several systems are used to check the effectiveness of the corrosion inhibitor by quantifying the corrosion rates of the coupons. The staff notes that loss of material and stress corrosion cracking are functions of their environment (chemistry, temperature, flow rate, stress, etc.), which should be taken into consideration when using test coupons.

Issue
It was not clear from the program basis documents that the corrosion coupons are exposed to a condition representative of the most detrimental environment for a given closed cycle system (highest temperature, stagnant conditions, etc.). In addition, it was not clear if the corrosion coupons would be stressed, which would be necessary to evaluate stress corrosion cracking.

Request
Justify how the corrosion coupons will represent a high susceptible material and environmental conditions observed in a Closed-Cycle Cooling Water System. Provide additional information regarding the adequacy of the corrosion coupons to verify the effectiveness of the program to minimize stress corrosion cracking.

B.2.1.12-6

Background
The SRP-LR states that past operating experience would not necessarily invalidate an aging management program because the feedback from operating experience should have resulted in appropriate program enhancements or new programs. A review of past operating experience indicated a recurring condition in the primary component cooling water system with loss of material in piping downstream of valves CC-V-444 (CR 05-04881) and CC-V-446 (CR 03-01549) apparently due to cavitation erosion from throttling. The applicant stated that it had conducted flow rebalancing to alleviate the concern.

Issue
It was not clear to the staff how the applicant has re-evaluated these areas after flow rebalancing was conducted to determine whether loss of material due to cavitation erosion remains an issue in the primary component cooling water system.

Request
Provide additional information on how the loss of material due to cavitation erosion was confirmed to have been eliminated or whether this remains an issue in the primary component cooling water system. If loss of material for this mechanism is still an applicable aging issue, provide information on what program is managing this aging effect and how.

RAI B.2.1.18-1

Background
The Updated Final Safety Analysis Report (UFSAR) Supplement description contained in the Standard Review Plan for License Renewal (SRP-LR), Table 3.3-2, "FSAR Supplement for Aging Management of Auxiliary Systems", provides an acceptable program description which includes the specific American Society for Testing and Materials (ASTM) International Standards to be used for the monitoring and controlling of fuel oil contamination to maintain fuel oil quality. License Renewal Application (LRA) A.2.1.18 "Fuel Oil Chemistry" states:

> ...New fuel oil is sampled and verified to meet the requirements of applicable American Society for Testing and Materials (ASTM) standards prior to offloading to the storage tanks. The program monitors fuel oil quality and the levels of water in the fuel oil which may cause the loss of material of the tank internal surfaces. The program monitors water and sediment contamination in diesel fuel...

Issue
Specifying the ASTM Standards to be used ensures that there is an adequate description of the critical elements of the Fuel Oil Chemistry Aging Management Program to provide assurance that the program will be properly executed during a period of extended operations.

Request

1. Justify the absence of specific ASTM Standards in your UFSAR Supplement provided in the LRA Appendix A. Alternatively, provide a revision to your UFSAR supplement to specify the specific ASTM standards used in your program.

RAI B.2.1.18-2

Background
GALL Report AMP XI.M30 Fuel Oil Chemistry, in the Generic Aging Lessons Learned (GALL) Report, states:

> **Scope of Program:** The program is focused on managing the conditions that cause general, pitting, and microbiologically-influenced corrosion (MIC) of the diesel fuel tank internal surfaces in accordance with the plant's technical specifications (i.e., NUREG-1430, NUREG-1431, NUREG-1432, NUREG-1433) on fuel oil purity and the guidelines of ASTM Standards D1796, D2276, D2709, D6217, and D4057...

ASTM Standards D2276-00, D2709-96 and D4057-95 are referenced at the end of section XI.M30.

In the LRA, the GALL Report AMP B.2.1.18 on Fuel Oil Chemistry states ASTM D2276, D2709 and D4057 are used in accordance with the GALL. The applicant's Technical Requirement Program 5.1 "Diesel Fuel Oil Testing Program", which provides controls for the required testing of both new fuel oil and stored fuel oil, references the use of ASTM D4057-81 and D2709-82 for the sampling of new fuel and ASTM D2276-06 and D4057-81 for the sampling of stored fuel.

<u>Issue</u>
The LRA is not consistent with the GALL in the fact that Technical Requirement 5.1, which governs the plant procedures used by the program, references different revisions of ASTM D4057, D2709 and 2276 than are listed in the GALL.

<u>Request</u>
Justify using different revisions of ASTM Standards D2709, D4057 and D2276 than those specified in the GALL in your Fuel Oil Chemistry Program. Alternatively, describe your plans to implement the GALL recommended versions of the ASTM Standards in question.

<u>RAI B.2.1.23-1</u>

<u>Background</u>
The "monitoring and trending" program element of GALL AMP XI.M35, "One-Time Inspection of ASME Code Class 1 Small-Bore Piping" states that a one-time volumetric inspection is an acceptable method for confirming the absence of cracking of ASME Code Class 1 small-bore piping. The GALL Report also states that the inspection of small bore piping should be performed at a sufficient number of locations to assure an adequate sample and that this number, or sample size, will be based on susceptibility, inspectability, dose considerations, operating experience, and limiting locations of the total population of ASME Code Class 1 small-bore piping locations. GALL Report AMP XI.M35 states that MRP-146 provides guidelines for identifying piping susceptible to one subset of cracking, including thermal stratification or turbulent penetrations. The applicant's program states that it will inspect for cracking in ASME Code Class 1 small-bore piping using qualified volumetric examination techniques, if available, and that if the non-destructive volumetric examination techniques have not been qualified, Seabrook Station will have the option to remove the weld for destructive examination. The applicant stated during the staff's audit that it will inspect 10% of the butt welds and 10% of the socket welds. In addition, the applicant stated that it may not inspect certain welds based on inaccessibility or high radiation exposure.

Issue

It is not clear to the staff if the applicant will either conduct an acceptable volumetric inspection or plan to do destructive examination. Based on the language in the applicant's program basis document, the staff noted that if an acceptable volumetric exam is not available before the period of extended operation, the applicant will have the option to perform destructive exams. In addition, the staff noted that the applicant proposed to inspect weld locations that are susceptible to SCC and cyclical loading, but the sampling methodology for the inspection was not presented. It was also not clear to the staff what part of the socket welds the applicant plans to inspect.

Request

1. Clarify and justify the use of destructive examination as an "option" within the program and FSAR supplement if an "acceptable" volumetric method isn't available.

2. Clarify what is meant by an "acceptable" volumetric inspection and justify the use of a volumetric technique if it is not consistent with GALL AMP XI.M35 recommendations.

3. Describe the methodology for choosing the types of welds to inspect and how this methodology will ensure the AMP adequately manages the effects of relevant forms of cracking during the period of extended operation.

4. Provide clarification on the methodology that will be used to manage inaccessible or high radiation exposure welds within the scope of the One-Time Inspection of ASME Code Class 1 Small-Bore Piping Program and justify this methodology.

5. Clarify the proposed examination volume approach for socket welds, and justify that the examination volume is sufficient and capable of detecting cracking in the subject socket welds.

RAI B.2.1.26-1

Background

The GALL Report AMP XI.M39 states that for components that do not have regular oil changes, tests for viscosity, neutralization number, and flash point may be used to determine lubricating oil suitability for continued use. In LRA AMP B.2.1.26, the applicant stated that Seabrook does not sample for flash point in lubricating oil samples. Instead, the applicant stated that when there is a potential for lubricating oil contamination by fuel, Seabrook will test the samples for fuel dilution. The applicant further stated that testing for fuel dilution is equivalent to testing for flash point because either test will provide an indication of fuel in-leakage.

Issue

The equivalency of the method the applicant uses to test for fuel dilution and the GALL Report AMP recommended flash point testing has not been substantiated.

Request

1. Discuss the method Seabrook uses to determine fuel dilution and how it compares to sampling for flash point (i.e., justify how this method is equivalent or conservative relative to flash point). In addition, provide the acceptance criteria for this method and corrective actions taken if a sample does not meet the acceptance criteria.

RAI B.2.2.2-1

Background
The Boral monitoring program is implemented to ensure that the aging effects of spent fuel pool neutron-absorbing material, which could compromise the criticality analysis, will be detected in the period of extended operation. The loss of material and the degradation of the neutron-absorbing material capacity are determined through testing of representative sample coupons. Such testing includes periodic verification of boron loss by performing areal density measurement of coupons and measurement of geometric changes in the coupons.

Issue
Strict control over the techniques used to prepare the coupons for testing as well as the associated tests performed are critical to obtaining accurate data that is used to perform trending analysis of the aging effects on the coupon. The staff requires more documentation of the applicant's operational experience with the testing of the coupons to determine if the program is able to perform as intended during the period of extended operation.

Request
Provide additional operational experience associated with the preparation of the coupons for testing, the performance of the associated tests, and the results of the tests.

RAI B.2.2.3-1

Background
For several AMR items related to nickel alloy components, the GALL Report recommends that aging be managed by a combination of approaches. These approaches include:

AMP XI.M2, Water Chemistry
AMP XI.M1, ASME Section XI, Inservice Inspections, Subsection IWB, IWC, and IWD
Compliance with NRC Orders
A commitment to implement NRC Bulletins and Generic letters
A commitment to implement industry guidelines

For the corresponding AMR items in the LRA, the LRA indicates that aging will be managed through the use of the Nickel Alloy Nozzles and Penetration AMP. In its review of this AMP, the staff found that the AMP was worded such that its implementation also implements the water chemistry and ASME Section XI AMPs. The AMP also contains language indicating compliance with NRC Orders.

In its review of this AMP and the FSAR, the staff found language which indicates that application of this AMP includes implementation of NRC Bulletins and Generic letters, and implementation of industry guidelines. This language was not, however, in the form of a commitment. The staff also noted that such a commitment was absent from the list of commitments provided in the LRA.

Issue
While the staff has little concern that this AMP would be applied without implementing NRC Bulletins, Generic Letters or industry guidelines, the staff fails to see how, in the absence of the commitments specified in the AMR items, implementation of the AMP, as written, is fully consistent with the AMR items.

Request
Please provide a commitment in the list of commitments to implement NRC Bulletins and Generic Letters, and to implement industry guidelines, or justify why such a commitment is not necessary.

RAI B.2.3.1-1

Background
The scope of the Metal Fatigue of Reactor Coolant Pressure Boundary Program includes both nuclear steam supply system (NSSS) and non-NSSS components and transients in UFSAR Section 3.9.1.1 that are required to be tracked. LRA Section 4.3 states that the metal fatigue TLAAs that are evaluated in the LRA fall into the following three categories:

Category (a) - Explicit fatigue analyses for 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.

Category (b) - 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.

Category (c) - 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.

In addition, LRA Section 4.3.1 states that the most limiting numbers of transients used in these NSSS component analyses are shown in Table 4.3.1-2, and are considered to be design limits. The staff confirmed that these transients are consistent with those listed in UFSAR Table 3.9(N)-1.

Issue
LRA Table 4.3.1-2 lists more plant design transients than those identified in Technical Specification (TS) 5.7 and TS Table 5.7-1. For example, in the TS table, normal condition transients include only plant heatup and shutdown; upset set transients include only loss of load w/o turbine roll, loss of all offsite power, partial loss of flow, and reactor trip from full power; faulted transients include large steam line break; and test transients include primary and secondary side hydrostatic test, and primary side leak test. It is not clear to the staff whether the design CUF fatigue analyses for NSSS pressure vessels and components were based on the design transients listed in TS Table 5.7-1 or the non-TS transients that were included in LRA Table 4.3.1-2 and UFSAR Table 3.9(N)-1.

In the event that a transient that is listed in LRA Table 4.3.1-2 but not in the TS occurs, it is not clear to the staff how the transient will be accounted for in accordance with the Metal Fatigue of Reactor Coolant Pressure Boundary Program during the period of extended operation.

The "parameters monitored/inspected" program element of GALL AMP X.M1 states the program monitors all plant transients that cause cyclic strains, which are significant contributors to the fatigue usage factor.

Request

(1) Clarify whether the Category (a) fatigue analysis and the Category (b) supplemental fatigue analysis were based on transients from TS Table 5.7-1 or LRA Table 4.3.1-2 [and in UFSAR Table 3.9(N)-1].

(2) Confirm that the plant-specific cycle counting procedure ensures those design transients that are listed in LRA Table 4.3.1-2 but not in TS 5.7 will be tracked and monitored (i.e., counted) in accordance with the Metal Fatigue of Reactor Coolant Pressure Boundary Program, during the period of extended operation. If these transients are not monitored during the period of extended operation, justify why they are not monitored by the Metal Fatigue of Reactor Coolant Pressure Boundary Program, consistent with the "parameters monitored/inspected" program element.

RAI B.2.3.1-2

Background
The scope of the Metal Fatigue of Reactor Coolant Pressure Boundary Program includes both NSSS and non-NSSS components and transients in UFSAR Section 3.9.1.1 that are required to be tracked. The applicant stated that the most limiting numbers of transients used in these NSSS component analyses are shown in LRA Table 4.3.1-2, and are considered to be design limits.

Issue
The staff noted that the transients are termed differently in the LRA, UFSAR, and relevant documents that were reviewed during the staff's audit. For example, upset condition transients such as "inadvertent startup of an inactive loop" or "inadvertent emergency core cooling system

actuation" are referred to differently in these documents. The staff also noted, during its audit, that the applicant's program basis document includes auxiliary transients such as "charging and letdown flow shutoff and return" or "letdown flow step decrease and return", however, these transients are not included in the list of design transients provided in LRA Table 4.3.1-2.

Request

(a) Justify that the difference of designations for the transients between LRA Table 4.3.1-2 and CUF analyses in the applicant's program basis document should not be aligned. Clarify and justify how the Metal Fatigue of Reactor Coolant Pressure Boundary Program and the associated on-site procedure will be capable of tracking transient occurrences to ensure that the design limit of 1.0 is not exceeded and that any assumptions that are made in the fatigue CUF analyses remain valid, if the designations for the transients are not consistent between the LRA, the UFSAR, and other relevant documents.

(b) Clarify the significance of the auxiliary transients used in fatigue CUF analyses, and explain how these transients are accounted for by the list of design transients provided in LRA Table 4.3.1-2.

RAI B.2.3.1-3

Background

The LRA Section B.2.3.1 and Commitment No. 41 states that the following enhancement will be made prior to entering the period of extended operation:

> The Metal Fatigue of Reactor Coolant Pressure Boundary Program will be enhanced to include additional transients beyond those defined in the TS and UFSAR.

Issue

The Metal Fatigue of Reactor Coolant Pressure Boundary Program does not identify these additional design transients that are monitored beyond those defined in the TS and UFSAR. The staff also noted that the applicant's program does not provide any description or the significance of these additional transients. The applicant's program also does not identify the components that these additional transients affect, specifically those CUF TLAAs in LRA Section 4.3 that the applicant dispositioned 10 CFR 54.21(c)(1)(iii)

Request

(a) Identify all additional design transients that are monitored by the Metal Fatigue of Reactor Coolant Pressure Boundary Program and describe why these additional transients are to be monitored. Discuss the significance of these additional transients to the TLAAs that have been identified in LRA Section 4.3.

(b) Clarify how these additional transients relate to the Technical Specification 5.7 and transients analyzed for in UFSAR Section 3.9.

(c) Clarify whether these transients were included in the new environmentally-assisted fatigue analysis evaluations that were prepared for license renewal in LRA Section 4.3.4.

If they were not included, provide justification why these transients are significant only for those analyses in the CLB and not significant for the analyses performed for the period of extended operation.

RAI B.2.3.1-4

Background
In LRA Section B.2.3.1, the applicant stated that the Metal Fatigue of Reactor Coolant Pressure Boundary Program monitors and tracks the number of transient cycles to ensure that the CUF

for select reactor coolant system components remains less than 1.0 through the period of extended operation. The applicant also stated the program ensured the environmental effect on fatigue sensitive locations are addressed. Locations with CUF approaching the design limit are reanalyzed, inspected, repaired, or replaced as necessary in accordance with applicable design codes. LRA Section B.2.3.1 states that pre-established action limits will permit completion of corrective actions before the design basis number of events is exceeded, and before the CUF, including environmental effects, exceeds the ASME Code limit of 1.0.

Issue
It is not clear to the staff if the Metal Fatigue of Reactor Coolant Pressure Boundary Program will perform cycle-counting, cycle-based fatigue monitoring, or stress-based fatigue monitoring for reactor coolant pressure boundary components (including the environmentally-assisted fatigue). The Metal Fatigue of Reactor Coolant Pressure Boundary Program does not provide details regarding the action limits that are set on design basis transient cycle counting activities or on CUF monitoring activities, or the corrective actions that will be implemented if an action limit of cycle counting or CUF monitoring is reached.

The staff has noted that LRA Section 4.3 sets a design limit of 1.0 for CUF analyses and environmentally-adjusted CUF analyses but the design limit for high energy line break locations is set to a value of 0.1. Furthermore, in order to maintain a design limit of 1.0, it should be noted that the action limit for cycle counting or CUF monitoring can be different if the same transient is used in a CUF analyse of a component or an environmentally-adjusted CUF analyses of another component.

Request

Define and justify the "action limit or limits" that will be used by the Metal Fatigue of Reactor Coolant Pressure Boundary Program for:

- design basis CUF values for Class 1 components and any non-Class 1 components evaluated to Class 1 component CUF requirements.

- environmentally-assisted CUF for the program's NUREG/CR-6260 equivalent or bounding locations, and, or

- Class 1 components that are within the scope of the applicant's high-energy line break analyses for Class 1 components.

RAI B.2.3.1-5

Background

The "corrective actions" program element in GALL AMP X.M1, "Metal Fatigue of Reactor Coolant Pressure Boundary" states that acceptable corrective actions include repair of the component, replacement of the component, and a more rigorous analysis of the component to demonstrate that the design code limit will not be exceeded during the extended period of operation.

Issue
In LRA Section B.2.3.1, the applicant stated that corrective actions may encompass one of several activities below:

1. Reanalyze affected component(s) for an increase in the number of that specific transient while accounting for other component-affecting plant transients that may be projected not to achieve their analyzed levels.

2. Perform a fracture mechanics evaluation of a postulated flaw in affected plant components, which, when coupled with an inservice inspection program, will serve to demonstrate flaw tolerant behavior.

3. Repair the affected component.

4. Replace the affected component.

Request
Provide a justification for the corrective action, to perform a fracture mechanics evaluation, which is not consistent with the recommendations of the "corrective actions" program element of GALL AMP X.M1

RAI B.2.3.1-6

Background
The "Detection of Aging Effects" program element in GALL AMP X.M1 ("Metal Fatigue of Reactor Coolant Pressure Boundary") states that the aging management program provides periodic update of the fatigue usage calculations. The LRA Section B.2.3.1 also states that the Metal Fatigue of Reactor Coolant Pressure Boundary Program will be enhanced to use a software program to count transients to monitor cumulative usage on selected components. The applicant also included this enhancement in LRA Commitment No. 42 in LRA Table A.3. LRA Section B.2.3.1 also states that "The program includes generation of a periodic fatigue monitoring report, including a listing of transient events, cycle summary event details, cumulative usage factors, a detailed fatigue analysis report, and a cycle projection report."

Issue

The staff noted that the LRA does not provide the details regarding the software package that will be used. It is not clear to the staff if the "program" being referred to is the Metal Fatigue of Reactor Coolant Pressure Boundary Program or the software program. It is also not clear to the staff if the software package will be used for cycle counting only or if it will also be used for cycle-based or stress-based fatigue analysis and includes periodic CUF updates.

Request

(a) Clarify, in detail, how the software package selected will be capable of monitoring those transients that are significant to fatigue usage such that the design limit of 1.0 is not exceeded during the period of extended operation, consistent with the recommendations in GALL AMP X.M1.

(b) Clarify how the software package will perform periodic CUF updates, consistent with the recommendations of the "detection of aging effects" program element of GALL AMP X.M1.

(c) Clarify how the software package referenced in LRA Section B.2.3.1 and Commitment No. 42 addresses and resolves the issue associated with NRC RIS 2008-30.

RAI 4.7.3-1

Background

The staff notes that in Section 4.7.3 of the LRA, the applicant stated that mechanical stress improvement process had been performed at one of the reactor vessel primary hot leg nozzle locations in 2009. To the staff, this indicates that materials susceptible to PWSCC (Alloys 600/82/182) are present in the system and that PWSCC either has occurred or is considered highly likely to occur at this location. The staff also notes that Section III.3 of Part 3.6.3 of the Standard Review Plan (NUREG 0800) states that in a leak before break evaluation the applicant must demonstrate that "PWSCC is not a potential source of pipe rupture". The staff further notes that Section III.7 of Part 3.6.3 of the Standard Review Plan states that "other regulatory guidance on LBB specifies that two mitigation methods are needed to address materials susceptible to an active stress corrosion cracking degradation mechanism".

Issue

Given that it appears that materials susceptible to PWSCC exist in the reactor coolant system (RCS) primary loop piping, given that there is a history of PWSCC in these materials, and given that it appears that not all of the susceptible materials have been mitigated, it is not clear to the staff that, in this case, the use of leak before break analyses during the period of extended operation are consistent with NRC guidance on the subject.

Request

Please: 1) specifically identify the location of all materials which are susceptible to PWSCC in the piping systems covered by leak before break analyses; 2) describe the inspection program covering these susceptible materials including inspections which monitor the growth of known indications; 3) describe in detail the results of these inspections including details on any known indications; 4) describe mitigation techniques which have been applied to these susceptible materials; and 5 a) demonstrate how the leak before break analyses are consistent with Section III.3 of Part 3.6.3 of the Standard Review Plan, i.e., that "PWSCC is not a potential

source of pipe rupture" or Section III.7 of Part 3.6.3 of the Standard Review Plan, i.e., that "two mitigation methods are needed to address materials susceptible to an active stress corrosion cracking degradation mechanism"; or 5 b) describe how activities described in 1) - 4) above and any other programs provide reasonable assurance that a LBB analysis, which considers the possibility of PWSCC, is bounded by the existing leak before break analysis; or 5 c) provide additional calculations which, considering the potential for PWSCC, demonstrate that the principles of LBB analyses are satisfied.

RAI 4.7.13-1

Background
In its evaluation of the second aspect of the applicant's assertion that evaluations conducted for LBB do not constitute a TLAA, the staff notes that the applicant refers to LRA Section 4.7.3. This section is a description of the LBB TLAA. In this section the applicant states that "the analyses involved with LBB are considered TLAAs." In this section the applicant also dispositions this TLAA in accordance with 10 CFR 54.21(c)(1)(i), indicating that the analyses remain valid for the period of extended operation. In its evaluation of the TLAA, the staff noted that the fatigue cycles used in the analysis were based on 40 years.

Issue
In light of the information presented in LRA Section 4.7.3, it appears to the staff that the LBB TLAA is, in fact, a TLAA which is based on the 40 year current operating term of the plant. It also appears to the staff that, in accordance with SRP-LR Paragraph 4.7.3.1.1 the applicant has identified an additional activity, cycle counting in this case, which allows the assumptions used in the original 40 year calculation to be confirmed for a 60 year plant life. The staff finds that this process allows the TLAA to be applied during the period of operation but that it does not constitute a new calculation based on 60 years. The staff therefore disagrees with the applicant's position that the LBB calculations do not constitute a TLAA.

Request
Please modify LRA Section 4.7.13 to indicate that the LBB calculations do constitute a TLAA which is addressed in LRA Section 4.7.3 or provide justification why these calculations do not constitute a TLAA.

December 14, 2010

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) – AGING MANAGEMENT PROGRAMS

Dear Mr. Freeman:

By letter dated May 25, 2010, NextEra Energy Seabrook, LLC (NextEra) submitted an application pursuant to Title 10 of the *Code of Federal Regulations* Part 54, to renew the Operating License NPF-86 for Seabrook Station, Unit 1 (Seabrook) 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 Paul Freeman from Richard Plasse dated December 14, 2010

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

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