January 13, 2011

SBK-L-11002
Docket No. 50-443

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

Seabrook Station
Response to Request for Additional Information
NextEra Energy Seabrook License Renewal Application
Aging Management Programs – Set 4

References:

1. NextEra Energy Seabrook, LLC letter SBK-L-10077, "Seabrook Station Application for Renewed Operating License," May 25, 2010. (Accession Number ML101590099)
2. NRC Letter "Request for Additional Information Related to the Review of the Seabrook Station License Renewal Application (TAC NO. ME4028) – Aging Management Programs" December 14, 2010 (Accession Number ML103260554)

In Reference 1, NextEra Energy Seabrook, LLC (NextEra) submitted an application for a renewed facility operating license for Seabrook Station Unit 1 in accordance with the Code of Federal Regulations, Title 10, Parts 50, 51, and 54.

In Reference 2, the NRC requested additional information in order to complete its review of the License Renewal Application (LRA). Enclosure 1 contains NextEra's response to the request for additional information and associated changes made to the LRA. For clarity, deleted LRA text is highlighted by strikethroughs and inserted texts highlighted by bold italics.

Commitment numbers 53, 54, 55, 56, 57, and 58 are added to the License Renewal Commitment List. There are no other new or revised regulatory commitments contained in this letter. A revised LRA Appendix A - Final Safety Report Supplement Table A.3, License Renewal Commitment List, has been included as attachment to NextEra Energy Seabrook, LLC letter SBK-L-11003, "Response to Request for Additional Information, Aging Management Programs – Set 5."

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NRR

If there are any questions or additional information is needed, please contact Mr. Richard R.Cliche, License Renewal Project Manager, at (603) 773-7003.

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

Sincerely,

NextEra Energy Seabrook, LLC.

Paul O. Freeman
Site Vice President

Enclosures:

Enclosure 1- Response to Request for Additional Information Seabrook Station License Renewal Application Aging Management Programs and Associated LRA Changes cc:

W.M. Dean,	NRC Region I Administrator
G. E. Miller,	NRC Project Manager, Project Directorate I-2
W. J. Raymond,	NRC Resident Inspector
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NEXTera ENERGY SEABROOK

I, Paul O. Freeman, Site Vice President of NextEra Energy Seabrook, LLC hereby affirm that the information and statements contained within are based on facts and circumstances which are true and accurate to the best of my knowledge and belief.

Sworn and Subscribed

Before me this

13 day of January, 2011

Paul O. Freeman
Site Vice President

Notary Public

MICHAEL D. O'KEEFE
MY COMMISSION EXPIRES FEB. 13, 2013
NOTARY PUBLIC
NEW HAMPSHIRE

Enclosure 1 to SBK-L-11002

Response to Request for Additional Information

Seabrook Station License Renewal Application

Aging Management Programs

and Associated LRA Changes

Request for Additional Information (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.

NextEra Energy Seabrook Response:

The Seabrook Station aging management program and Section B.2.1.1 of the license renewal application have been revised to state that Risk Informed ISI is implemented as an alternative to categories B-F and B-J, as approved by the NRC, for each individual ISI interval, and, should Risk Informed ISI not be approved during the period of extended operation, Seabrook Station will follow the applicable requirements of ASME Section XI, Subsection IWB.

Based on the above discussion, the following change has been made to the Seabrook Station License Renewal Application:

1. In Section B.2.1.1, on page B-17, the 5th paragraph is revised as follows:

   "Examination Categories B-P and C-H, require VT 2 visual examination (IWA-5240) during system leakage testing of all pressure retaining ASME Code Class 1 and 2 components, according to Tables IWB-2500-1 and IWC-2500-1 respectively. The extent and schedule of inspections, in accordance with Tables IWB-2500-1 and IWC-2500-1 ensure detection of aging degradation before the loss of the intended function. The Seabrook Station ASME Section XI Inservice Inspection Program performs the necessary inspections per the requirements of tables IWB-2500-1, IWC-2500-1, and IWD-2500-1. These inspections include the applicable portions of examination categories; B-A, B-B, B-D, B-G-1, B-G-2, B-K, B-L-2, B-M-2, B-N-1, B-N-2, B-N-3 B-O, B-P, C-A, C-B, C-C, C-F-1, C-F-2, C-H, D-A, and D-B. The applicable portions of categories B-F and B-J are currently included in the Risk Informed Inservice Inspection Program. ***Risk Informed ISI is implemented as an alternative to categories B-F and B-J, as approved by the NRC, for each individual ISI interval. Should Risk Informed ISI not be approved during the period of extended operation, Seabrook Station will follow the applicable requirements of ASME Section XI, Subsections IWB.*** Examination categories not listed above are not applicable to Seabrook Station."

Request for Additional Information (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.

NextEra Energy Seabrook Response:

1) The yield strength of the Seabrook Station Unit 1 reactor head closure studs material does not exceed 150 ksi. The documentation of the test results for the Unit 1 studs is contained in UFSAR Table 5.3-4. As stated in the LRA, during refueling outage 7 (Fall of 2000), the reactor head closure studs were replaced with the PlasmaBond coated Unit 2 studs. As part of a periodic replacement program (e.g. after the PlasmaBond coating outlives its useful anti-galling life), the Unit 2 reactor head closure studs were removed and replaced with the Plasma Bond coated Unit 1 studs during refueling outage 13 (Fall of 2009). These Unit 2 studs are presently in storage. Review of the Unit 2 CMTRs revealed that the yield strength from the test coupon for one of the bars from which the studs were manufactured from was measured at 151.75 ksi. Prior to entering the period of extended operation, the stud(s) manufactured from this bar will be replaced with ones that do not exceed 150 ksi measured yield strength.

   Based on the above discussion, the following changes have been made to the Seabrook Station License Renewal Application:

   1. In Section B.2.1.3, on page B-30, revised the last paragraph as follows:

      Seabrook Station implements the guidance outlined in RG 1.65 ***Rev 1*** for preventive measures. These preventive measures include material selection and use of appropriate coatings and lubricants. Seabrook Station has 54 reactor head closure studs and 54 spare studs. All are manufactured from SA-540, Class 3, Grade B24 material (UFSAR Table 5.2-2). The maximum tensile strength is less than 170 ksi (UFSAR Section 1.8) ***and the yield strength is less than 150* ksi *(UFSAR Table 5.3-4)***. The reactor head closure studs are coated with an anti-galling compound (PlasmaBond) and a station approved lubricant is utilized during installation/removal of the studs that do not contain molybdenum disulfide (MoS_2).

   2. In section A.3, on page A-43, added the following commitment to the License Renewal Commitment List:

No.	***PROGRAM or TOPIC***	***COMMITMENT***	***UFSAR LOCATION***	***SCHEDULE***
53	***Reactor Head Closure Studs***	***Replace the spare reactor head closure stud(s) manufactured from the bar that has a yield strength > 150 ksi with ones that do not exceed 150 ksi.***	*A.2.1.3*	***Prior to the period of extended operation.***

2) The Seabrook Station Reactor Head Closure Studs aging management program has provisions to preclude the use of material with yield strength greater than 150 ksi. As stated above, the Reactor Heads Closure Studs program will implement the guidance outlined in Regulatory Guide (RG) 1.65 Rev 1, "Material and Inspection for Reactor Vessel Closure Studs", for preventive measures. These preventive measures include material selection, appropriate coatings, and lubricants. The regulatory position stated in the April 2010 revision 1 to RG 1.65 relays the NRC staff position that the measured yield strength of stud material should not exceed 1034 Mpa (150 ksi). Since the Seabrook Station Reactor Head Closure Studs program will implement the guidance of RG 1.65 Rev 1, Seabrook Station aligns with the regulatory position that the yield strength for the stud material should not exceed 1034 Mpa (150 ksi).

Request for Additional Information (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.

NextEra Energy Seabrook Response:

1) Operating experience No. 2 described in LRA Section B.2.1.3 is a plant condition report from 2002, which identified slight coating of rust and discoloration on the PlasmaBond™ coating on the reactor head closure studs. The rust was limited to a small area near the top of the threaded portion of the stud that threads into the reactor vessel flange where the stud contacts the alloy steel reactor vessel stud hole. The inspection of the studs revealed no thread damage and the slight coating of rust and discoloration on the PlasmaBond™ coating was determined not to be an issue. The PlasmaBond™ vendor indicates that the discoloration was due to tarnishing of the silver in the coating.

   The reactor vessel head studs are an alloy steel. During refueling outages when the studs are not exposed to the normal operating temperatures of approximately 500°F, preventive measures are taken during their cleaning and re-installation to prevent potential corrosion of the stud material. The threaded portions of the studs are protected by the Plasma Bond™ coating and require no additional protection however, as a precaution; the studs are coated with WD-40 lubricant after cleaning and again prior to installation.

   The reactor vessel stud holes are machined and threaded directly into the reactor vessel flange which is also an alloy steel. Again, during refueling outages when the reactor head closure studs are removed, the stud holes are cleaned to remove any particles or deposits that may have accumulated in the stud hole. WD-40 lubricant is used to aid in stud installation and to clean and later protect the reactor vessel stud hole threads as an added measure to prevent potential corrosion of the stud material. The slight coating of rust identified in 2002 appears to have come from the reactor vessel stud hole.

   Based on the above discussion, this small amount of surface rust and discoloration on the PlasmaBond™ coating is not considered an aging effect that requires management during the extended period of operation.

2) The reactor vessel studs were coated with a thin film of a highly adherent, nickel-silver/palladium metallic alloy applied by the PlasmaBond™ process to prevent galling of the alloy steel studs, nuts and stud hole threads in the reactor vessel flange. The PlasmaBond coating is stable at the design and operating temperatures of the reactor vessel and is compatible with the stud, nut and vessel flange materials and the operating environment. The PlasmaBond coating is applied to prevent galling and assure successful installation and subsequent removal of the reactor vessel closure head studs and nuts.

WD-40 is a water displacing lubricant applied to the reactor vessel studs prior to installation as an additional aid to facilitate installation and cleaning of the studs. Based on the manufacturer's technical data sheet, WD-40 has an operating temperature range from (-) 10°F to (+) 200°F. The operating temperature of the reactor vessel closure head studs is estimated to approach 500°F. According to the manufacturer, any WD-40 exposed to reactor vessel metal temperature at operating condition would carbonize. Any carbonized deposits would have no adverse effect on the PlasmaBond coating or reactor vessel stud or flange materials. The chemical composition of WD-40 is considered by the manufacturer as proprietary information, but the manufacturer did verify that WD-40 lubricant does not contain any molybdenum disulfide. WD-40 has also been approved for use on the external surfaces of all equipment as a chemical/expendable product under the Seabrook Station expendable products program.

WD-40 lubricant is presently and has been used for the installation and cleaning of the reactor vessel closure head studs without any indication that the WD-40 is a cause of corrosion. Discussions with the Seabrook Station personnel responsible for removal and cleaning of the reactor vessel closure head studs indicate that no evidence of corrosion has been found attributable to the use of WD-40. The expendable products program approval of and the plant specific operating experience with the use of WD-40 lubricant on the reactor vessel closure head studs shows that the WD-40 does not cause any type of corrosion and is compatible with the vessel materials and surrounding service environment.

Request for Additional Information (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.

NextEra Energy Seabrook Response:

1) The reactor vessel internals that are fabricated from Alloy X-750 are the clevis insert bolts. These clevis insert bolts are identified in the UFSAR Table 5.2-4 as Inconel X-750 SA 637, Gr. 688, Type II.

2) The plant-specific experience associated with PWR vessel internal components fabricated from Alloy X-750 is as follows:

   Seabrook Station control rod drive mechanism (CRDM) guide tube support pins are used to align the bottom of the guide tube assembly into the top of the upper core plate. Two support pins are locked into place in the bottom flange of each of the fifty-seven guide tube assemblies and are inserted into the upper core plate to provide lateral support. The support pins were fabricated from Inconel X-750 material which had been shown by industry operating experience to be susceptible to Primary Water Stress Corrosion Cracking (PWSCC). Subsequently, Westinghouse developed a new stainless steel support pin design and fabrication technique that demonstrated no susceptibility to PWSCC. The redesigned support pins are fabricated from cold worked stainless steel material. During Refueling Outage 11 (October of 2006), all of the CRDM guide tube support pins, a total of 114, were replaced with the cold worked stainless steel support pins.

3) The future plans for managing age-related degradation for the clevis insert bolts will be determined by the implementation of MRP-227, "Material Reliability Program: Pressurized Water Reactor Internals Inspection and Evaluation Guidelines," and the ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD Program requirements.

Request for Additional Information (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.

NextEra Energy Seabrook Response:

Common Maintenance and Engineering department practices at Seabrook Station include notification of the FAC Engineer whenever a system or component is opened for FAC related work or inspection. The FAC Engineer also follows all such work or inspection to the extent practical to ensure that the scope of the condition is understood. These common practices ensure that when significant degradation is noted downstream of a component (such as a valve or nozzle), appropriate consideration is given to the possible effect on upstream components. However, this is not explicitly called for in the Flow Accelerated Corrosion Program.

1) The Flow Accelerated Corrosion Program is revised to state that if significant wear is detected in piping immediately downstream of a valve, orifice, equipment nozzle or other like component, the component should also be examined. Examination of these components will be by ultrasonic, radiographic, and/or visual techniques typically utilized to inspect valves, orifices and equipment nozzles. This change is consistent with the program requirements of NSAC-202L-R2, "Recommendations for an Effective Flow Accelerated Corrosion Program."

2) The Flow Accelerated Corrosion program is revised to state that should follow-up inspection of a valve, orifice, equipment nozzle or other like component be required as the result of finding significant wear in the downstream piping, such follow-up inspection will be implemented in accordance with the Seabrook Station corrective action program.

Based on the above discussion, the following change has been made to the Seabrook Station License Renewal Application:

1. In Section B.2.1.8, on page 52, the 6th paragraph is revised as follows:

   "Components are inspected for wall thinning due to flow-accelerated corrosion using ultrasonic or radiography examinations. Ultrasonic examination provides more complete data for measuring the remaining wall thickness. As described in the EPRI Recommendations for an Effective Flow Accelerated Corrosion Program, radiography is commonly used on small-bore piping because it can be performed without removing pipe insulation and during plant operation with components in service. Evaluation of the results is performed by FAC engineers and is not used to identify other mechanisms (i.e., cracking or weld indications). Valves, orifices, equipment nozzles, and other like components that cannot be inspected completely with ultrasonic examinations due to their shape and thickness are evaluated based on the wear of piping located immediately downstream. ***If significant wear is detected in piping immediately downstream of a valve, orifice, equipment nozzle or other like component, the component should also be examined. Examination of these components will be by ultrasonic, radiographic, and/or visual techniques typically utilized to inspect valves, orifices and equipment nozzles. Should follow-up inspection of a valve, orifice, equipment nozzle or other like component be required as the result of finding significant wear in the downstream piping, such follow-up inspection will be implemented in accordance with the Seabrook Station corrective action program.*** Analytical models developed with computer programs, including CHECWORKS, are used to predict locations that are susceptible to flow-accelerated corrosion in piping systems based on specific plant data including material, configuration, hydrodynamic conditions, and operating conditions."

Request for Additional Information (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.

NextEra Energy Seabrook Response:

1. Based on the approved alternate repair criteria, the Seabrook Station steam generator tube-to-tubesheet welds are not included in the reactor coolant pressure boundary. This alternate repair criteria has not yet been permanently approved.

2. Unless an alternate repair criteria changing the ASME code boundary is permanently approved by the NRC, or the Seabrook Station steam generators are replaced to eliminate PWSCC-susceptible tube-to-tubesheet welds, Seabrook Station will submit a plant-specific aging management program to manage the potential aging effect of cracking due to PWSCC at least twenty-four months prior to entering the Period of Extended Operation.

This plant-specific program will either

1. Perform a one-time inspection of a representative sample of tube to tubesheet welds in all steam generators to determine if PWSCC cracking is present and, if cracking is identified, resolve the condition through engineering evaluation justifying continued operation or repair the condition, as appropriate, or

2. Perform an analytical evaluation showing that the structural integrity of the steam generator tube-to-tubesheet interface is adequately maintaining the pressure boundary in the presence of tube-to-tubesheet weld cracking, and that the tube-to-tubesheet welds are not required to perform a reactor coolant pressure boundary function.

Based on the above discussion, the following change has been made to the Seabrook Station License Renewal Application:

1. The following commitment is added to Section A.3 LICENSE RENEWAL COMMITMENT LIST of the License Renewal Application on page A-43:

No.	PROGRAM or TOPIC	COMMITMENT	UFSAR LOCATION	SCHEDULE
54	***Steam Generator Tube Integrity***	***Unless an alternate repair criteria changing the ASME code boundary is permanently approved by the NRC, or the Seabrook Station steam generators are changed to eliminate PWSCC-susceptible tube-to-tubesheet welds, submit a plant-specific aging management program to manage the potential aging effect of cracking due to PWSCC at least twenty-four months prior to entering the Period of Extended Operation.***	***A.2.1.10***	***Program to be submitted to NRC at least 24 months prior to the period of extended operation.***

Request for Additional Information (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.

NextEra Energy Seabrook Response:

1. Seabrook Station Westinghouse Model F steam generator divider plate and weld materials are Inconel (ASME-SB-168) Alloy 600/82/182.

2. Seabrook Station will perform an inspection of each steam generator prior to entering the period of extended operation to assess the condition of the divider plate assembly unless operating experience and/or analytical results show that crack propagation into RCS pressure boundary is not possible, then the inspections need not be performed. The inspection techniques used will be capable of detecting primary water stress corrosion cracking in the steam generator divider plate assemblies and their associated welds. Any evidence of cracking will be documented and evaluated under the corrective action program.

   Seabrook Station remains involved with the on-going industry studies conducted by EPRI related to divider plate cracking. This participation will ensure that any inspection requirements or other resolution actions recommended to the industry are evaluated and implemented as appropriate.

Based on the above discussion, the following change has been made to the Seabrook Station License Renewal Application:

1. The following commitment is added to Section A.3 LICENSE RENEWAL COMMITMENT LIST of the License Renewal Application on page A-43:

No.	PROGRAM OR TOPIC	COMMITMENT	UFSAR LOCATION	SCHEDULE
55	***Steam Generator Tube Integrity***	***Seabrook will perform an inspection of each steam generator to assess the condition of the divider plate assembly unless operating experience and/or analytical results show that crack propagation into RCS pressure boundary is not possible, then the inspections need not be performed.***	***A.2.1.10***	***Prior to entering the period of extended operation***

Request for Additional Information (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.

NextEra Energy Seabrook Response:

The Open-Cycle Cooling Water System program discusses visual inspection which is performed typically by divers during refuel outages. Such visual inspection would provide limited ability to determine hardening and loss of strength due to elastomer degradation. LRA Section B.2.1.11 is revised to include physical or manual manipulation of elastomers to detect hardening and loss of strength due to elastomer degradation.

Based on the above discussion, the following changes have been made to the Seabrook Station License Renewal Application:

1. In Section B.2.1.11, on page B-68, the 4th paragraph is revised as follows:

   The Seabrook Station Open-Cycle Cooling Water System Program includes a variety of inspection and testing methods such as visual, eddy current and ultrasonic test (UT) inspections on plant heat exchangers and piping These activities are designed to detect degradation due to corrosion, microbiologically influenced corrosion, biofouling, silt, debris, and scaling prior to loss of intended function. Visual inspections of elastomers (e.g., rubber expansion joints) are performed to detect erosion~~ and elastomer degradation~~. Rubber expansion joints are also examined when removed for maintenance activities to provide a more detailed evaluation of condition***, including detection of hardening and loss of strength due to elastomer degradation***.

2. In Section B.2.1.11, on page B-68, a new paragraph is added following the 4th paragraph as follows:

    The program monitors the internal surfaces of elastomeric components for hardening and loss of strength, cracking, and for loss of material due to wear by visual inspection and manual or physical manipulation when the internal surfaces are accessible during periodic surveillances or maintenance activities.

Request for Additional Information (RAI) 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.

NextEra Energy Seabrook Response:

The Thermal Barrier Cooling System in located entirely inside of containment. The higher limit to the hydrazine operating range was established to minimize radiation exposure required for hydrazine makeup during power operations. Action Levels 1 and 2 (<5 ppm and <1 ppm, respectively) remained consistent with those specified by the latest revision of the EPRI guidelines; however the operating limit of hydrazine was increased from 200 ppm to 300 ppm. At the same time, the Thermal Barrier system sulfate limit was raised to 500 ppb. By January of 2010, Seabrook Station had returned the normal operating hydrazine concentration in the Thermal Barrier system to less than 150 ppm and the normal operating sulfate concentration in the Thermal Barrier system to approximately 100 ppb. Hydrazine and sulfate have remained at these levels since that time.

In light of the current operating levels for hydrazine and sulfates in the Thermal Barrier system, the exceptions regarding these elevated levels and the inconsistency with the EPRI Guidelines are withdrawn. The License Renewal Application, Appendix B, is revised accordingly.

In Section B.2.1.12, on page 77, sub-paragraph a is deleted.

In Section B.2.1.12, on page 78, sub-paragraph b is deleted.

In Section B.2.1.12, on page 78, sub-paragraph c is renumbered as “a”.

In Section B.2.1.12, on page 79, Exception 2 is deleted.

In Section B.2.1.12, on page 80, Exception 3 is deleted.

In Section B.2.1.12, on page 81, Exception 4 is renumbered as “Exception 2”.

In Section B.2.1.12, on page 82, Exception 5 is renumbered as “Exception 3”.

A condition report has been generated to ensure that the station program documents reflect the EPRI Guideline operating ranges and Action Level values for hydrazine and sulfates.

Seabrook Station evaluated the significance of allowing operation of the system within the elevated ranges of hydrazine and sulfates concentrations prior to initiating the revised operating ranges and determined it to be acceptable. Routine monitoring during operation at the elevated ranges showed no indication of system or component degradation.

The following commitment is added to Appendix A, Section A.3:

No.	PROGRAM or TOPIC	COMMITMENT	UFSAR LOCATION	SCHEDULE
56	***Closed-Cycle Cooling Water System***	***Revise the station program documents to reflect the EPRI Guideline operating ranges and Action Level values for hydrazine and sulfates.***	*A.2.1.12*	***Prior to entering the period of extended operation.***

Request for Additional Information (RAI) 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.

NextEra Energy Seabrook Response:

A condition report has been generated to revise the on-site guidelines for the Diesel Generator Cooling Water Jacket pH Action Levels to be consistent with the EPRI Closed Cooling Water Chemistry Guideline Report. This revision will include addition of a specified Action Level 1 pH limit in addition to an Action Level 2 limit.

The following commitment is added to Appendix A, Section A.3:

No.	PROGRAM or TOPIC	COMMITMENT	UFSAR LOCATION	SCHEDULE
57	***Closed-Cycle Cooling Water System***	***Revise the station program documents to reflect the EPRI Guideline operating ranges and Action Level values for Diesel Generator Cooling Water Jacket pH.***	***A.2.1.12***	***Prior to entering the period of extended operation.***

Request for Additional Information (RAI) 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.

NextEra Energy Seabrook Response:

Maintenance Rule activities are not being credited to manage the aging effects in the closed-cycle cooling water system during the period of extended operation.

The License Renewal Application Section B.2.1.12 has been revised to remove reference to the Maintenance Rule and state that EPRI TR-1007820, "Closed Cooling Water Chemistry," notes that performance monitoring is typically part of an engineering program that verifies the component active functions. Performance and functional testing is not included as a part of the Seabrook Station Closed-Cycle Cooling Water System Program.

Based on the above discussion, the following change has been made to the Seabrook Station License Renewal Application:

1. In Section B.2.1.12, on page B-82, Exception 5, Justification for Exception, is revised as follows:

   EPRI 1007820 notes that performance ~~testing~~ ***monitoring*** is typically part of an engineering program that verifies the component active functions. ~~These activities would fall under the Maintenance Rule (10 CFR 50.65). This being the case, p~~***P***erformance and functional testing is not included as a part of the Seabrook Station Closed-Cycle Cooling Water System Program. Seabrook Station uses corrosion monitoring and internal inspections of opportunity to monitor program effectiveness at managing component degradation that could impact a passive function. Corrosion monitoring is accomplished through trending of the normal plant periodic sampling, and monitoring of corrosion coupons. The periodic sampling tests for corrosion products which when trended will give an indication of the rate of corrosion ongoing in a system. Seabrook Station also places test coupons in the Primary Component Cooling Water system and Thermal Barrier Cooling Water System to check the effectiveness of the corrosion inhibitor by quantifying the corrosion rates of the coupons. Seabrook Station Chemistry procedure, *"Visual Inspection Format for Plant Components"* provides instructions for visual inspection of individual components (e.g., valves, pumps, piping segments, heat exchangers) looking for pitting, general corrosion film presence, biological activity, deposits, etc. This procedure is used to monitor for corrosion in the component cooling water systems when the system or components are opened for maintenance.

Request for Additional Information (RAI) 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.

NextEra Energy Seabrook Response:

The nuclear industry has collected information on how to identify that an aging effect is occurring and has developed programs to train plant staff on how to correlate the observed condition to a possible aging effect. Inspections completed under the implemented license renewal aging management programs will require training and qualification of the personnel completing the inspections ensuring that these inspections will adequately address potential aging effects and associated acceptance criteria on the in-scope materials.

The results of internal visual inspections are evaluated to identify any adverse or potentially adverse conditions. Such conditions are documented, evaluated and corrected in accordance with the Seabrook Station corrective action program. System components must meet system design requirements, such as minimum wall thickness. Corrosion coupons are monitored and evaluated to ensure that material corrosion rates will not result in violation of system design requirements.

Based on the above discussion, the following change has been made to the Seabrook Station License Renewal Application to reference the appropriate training required for inspections under this program:

1. In Section B.2.1.12, on page B-78, added the following new paragraph after the last paragraph.

   Personnel completing inspections under the License Renewal Aging Management Close-Cycle Cooling Water System Program will be trained to identify the inspection parameters associated with the aging effects monitored by the program. As an example, the industry has developed the "Identification and Detection of Aging Issues" (EPRI 1007932) training program which is supplemented with an Aging Assessment Field Guide (EPRI 1007933) and Aging Identification and Assessment Checklists for Mechanical Components (EPRI 1009743). The EPRI training modules and/or current industry endorsed training program at the time of implementation would serve as the basis for the training and qualification program.

Request for Additional Information (RAI) 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.

NextEra Energy Seabrook Response:

Corrosion coupon monitoring will be used to assess the effectiveness of corrosion inhibitors by quantifying the corrosion rates of the coupons. The current location of the coupons in the closed-cycle cooling water system is appropriate for monitoring for effectiveness of corrosion inhibitors.

The corrosion coupons are not pre-stressed and are not used to monitor for stress corrosion cracking. Evaluation of stress corrosion cracking (SCC) in components managed by the Closed-Cycle Cooling Water Program is done by visual inspection of the applicable components when the components are open for maintenance or other activity. Components within the scope of this program that are susceptible to SCC will be specifically addressed in the Seabrook Station procedure for visual inspection of individual components for evidence of pitting, general corrosion film presence, biological activity, deposits, stress corrosion cracking, etc. This procedure is used by the Chemistry department to monitor the closed-cycle cooling water systems when the components are opened for maintenance.

Based on the above discussion, the Chemistry Procedure is being changed and tracked in the corrective action program, and the following change has been made to the Seabrook Station LRA:

1. In Section B.2.1.12, page B-78, the last paragraph is revised as follows:

   The Seabrook Station Closed-Cycle Cooling Water System Program recognizes that component inspections are an important part of an overall chemistry program to assess corrosion control and chemistry control effectiveness. Seabrook Station Chemistry Procedure, *"Visual Inspection Format for Plant Components"* provides instructions for visual inspection of individual components (e.g., valves, pumps, piping segments, heat exchangers) looking for pitting, general corrosion film presence, biological activity, deposits, ***stress corrosion cracking,*** etc. when a system or component is open for maintenance. Historical records of these inspections, including photographs, are maintained for comparative purposes.

2. In Section B.2.1.12, on page B-82, the last paragraph is revised to read as follows:

   EPRI 1007820 notes that performance testing is typically part of an engineering program that verifies the component active functions. These activities would fall under the Maintenance Rule (10 CFR 50.65). This being the case, performance and functional testing is not included as a part of the Seabrook Station Closed-Cycle Cooling Water System Program. Seabrook Station uses corrosion monitoring and internal inspections of opportunity to monitor program effectiveness at managing component degradation that could impact a passive function. Corrosion monitoring is accomplished through trending of the normal plant periodic sampling, and monitoring of corrosion coupons. The periodic sampling tests for corrosion products which when trended will give an indication of the rate of corrosion ongoing in a system. Seabrook Station also places test coupons in the Primary Component Cooling Water system and Thermal Barrier Cooling Water System to check the effectiveness of the corrosion inhibitor by quantifying the corrosion rates of the coupons. Seabrook Station Chemistry procedure, *"Visual Inspection Format for Plant Components"* provides instructions for visual inspection of individual components (e.g., valves, pumps, piping segments, heat exchangers) looking for pitting, general corrosion film presence, biological activity, deposits, ***stress corrosion cracking,*** etc. This procedure is used to monitor for corrosion in the component cooling water systems when the system or components are opened for maintenance.

Request for Additional Information (RAI) 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.

NextEra Energy Seabrook Response:

Flow re-balancing of the system performed in 2003 was expected to eliminate the cavitation-induced wear downstream of the throttled butterfly valves in question, and therefore, no follow-up inspections were scheduled as part of the corrective action process.

To ascertain whether the loss of material due to cavitation erosion has been eliminated or whether this remains an issue in the primary component cooling water system, the piping downstream of these valves will be inspected for loss of material prior to entering the period of extended operation. This inspection will be performed during the ten year period prior to the period of extended operation to allow adequate time for the condition to re-appear following the prior repairs.

A condition report has been initiated to perform these inspections during the ten year period prior to entering the period of extended operation.

Request for Additional Information (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.

NextEra Energy Seabrook Response:

Section A.2.1.18 should have listed the specific ASTM standards that monitor the new fuel oil and the stored fuel oil at Seabrook Station. The following change has been made to the Seabrook Station License Renewal Application:

1. In Section A.2.1.18, on Page A-12, revised the 2nd paragraph as follows:

   New fuel oil is sampled and verified to meet the requirements of applicable American Society for Testing and Materials (ASTM) standards ***D4057 and D2709*** prior to offloading to the storage tanks. ***Stored fuel oil is sampled and verified to meet the requirements of ASTM D2276 or ASTM D4057, and ASTM D2709.*** 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.

Request for Additional Information (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.

Issue

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.

Request

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.

NextEra Energy Seabrook Response:

Prior to the period of extended operation, Seabrook Station Technical Requirement Program 5.1 (Diesel Fuel Oil Testing Program) will be updated to the revision levels listed in GALL XI.M30 Rev 1 for ASTM standards ASTM D2709 and ASTM D4057. The revision levels referenced in GALL XI.M30 Rev 1 for these standards are ASTM D2709-96 and ASTM D4057-95.

Seabrook Station will take an exception to the Fuel Oil Chemistry Program related to ASTM standard D2276. Seabrook Station utilizes revision 6 of ASTM D2276 (ASTM D2276-06) instead of revision 0 as listed in GALL Rev. 1 (ASTM D2276-00). The basic methodology for determining particulate has not been changed from revision 0 to 6. This standard has been updated to include better figures and incorporating notes from previous revisions into the procedure steps. The figure and methodology for taking aviation jet fuel samples has changed but not used to obtain the diesel fuel oil samples.

Based on the above discussion, the following changes have been made to the Seabrook Station License Renewal Application:

1. In Section B.2.1.18, on page B-110, added a new exception to NUREG-1801 as follows:

   4. NUREG-1801 XI.M30 Rev 1, references ASTM Standard D2276-00 for determining particulate in diesel fuel oil. Seabrook uses ASTM Standard D2276-06

   Justification for the Exception

   The basic methodology for determining particulate has not been changed from ASTM D2276-00 to ASTM D2276-06. The standard has been updated to include better figures

and incorporating notes from previous revisions into the procedure steps. The figure and methodology for taking aviation jet fuel samples has changed but not used at Seabrook Station to obtain the diesel fuel oil samples.

Program Elements Affected: Parameters Monitored or Inspected (Element 3) and Acceptance Criteria (Element 6).

2. In Section B.2.1.18, on page 111, added a new enhancement as follows:

   5. Update Technical Requirement Program 5.1 (Diesel Fuel Oil Testing Program) ASTM standards to ASTM D2709-96 and ASTM D4057-95 required by the GALL XI.M30 Rev 1 prior to the period of extended operation.

3. In Appendix A, added a new commitment as follows:

No.	*PROGRAM or TOPIC*	*COMMITMENT*	*UFSAR LOCATION*	*SCHEDULE.*
58	*Fuel Oil Chemistry*	*Update Technical Requirement Program 5.1, (Diesel Fuel Oil Testing Program) ASTM standards to ASTM D2709-96 and ASTM D4057-95 required by the GALL XI.M30 Rev 1*	*A.2.1.18*	*Prior to the period of extended operation.*

Request for Additional Information (RAI) B.2.1.23-1

Background

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.

NextEra Energy Seabrook Response:

1) To clarify the use of "optional" methods, Section A.2.1.23 and Section B.2.1.23 of the License Renewal Application have been changed to specify that if no demonstrated method of non-destructive volumetric examination capable of detecting cracking in socket welded piping is available, Seabrook Station will remove the selected weld(s) for destructive examination. The following changes have been made to the Seabrook Station License Renewal Application:

   1. In Section A.2.1.23, on page A-14, the 2nd paragraph is revised as follows:

      The inspection sample determination will include both socket welds and butt welds. ~~If non-destructive volumetric inspection techniques have not been qualified, Seabrook will have the option to remove the weld for destructive examination.~~***If no demonstrated***

method of non-destructive volumetric examination capable of detecting cracking in socket welded piping is available, Seabrook Station will remove the selected weld(s) for destructive examination.

2. In Section B.2.1.23, on page B-130, the last sentence in the 1st paragraph is revised as follows:

   ~~If non-destructive volumetric inspection techniques have not been qualified, Seabrook will have the option to remove the weld for destructive examination.~~***If no demonstrated method of non-destructive volumetric examination capable of detecting cracking in socket welded piping is available, Seabrook Station will remove the selected weld(s) for destructive examination.***

2) Seabrook Station does not refer to "acceptable" volumetric inspection techniques but uses the term "qualified" in discussion of inspection techniques. In reference to volumetric inspection of socket welded fittings, the term "qualified" is changed to "demonstrated" in that volumetric techniques must be demonstrated to be capable of detecting cracking. The term "demonstrated" was specifically used to indicate processes that have been, or may be developed to address this particular condition. Appendix B will be revised to clarify the use of "demonstrated" versus "qualified". The following change has been made to the Seabrook Station License Renewal Application:

   1. In Section B.2.1.23, page B-131, the 1st paragraph in Operating Experience is revised as follows:

      The Seabrook Station One-Time Inspection of ASME Code Class 1 Small Bore Piping Program is a new program. Both plant and industry operating experience will be used to establish the program and to ensure that this inspection uses volumetric inspection techniques with demonstrated capability ~~and a proven industry record~~ to detect cracking in piping weld~~s and base material. The specific examination techniques utilized will be qualified prior to performing the examinations~~.

3) Approximately 450 ASME Code Class 1 small-bore welds have been identified as the population of welds within the scope of this program. Approximately 150 of these are socket welds. The sample selection will consist of 10% of the welds within this population (approximately 45) and include at least 10% of the total number of socket welds in this population (approximately 15). The criteria of a 10% sample size which includes 10% of the in-scope socket welds will be added to Appendix B of the License Renewal Application. The following change has been made to the Seabrook Station License Renewal Application:

   1. In Section B.2.1.23, page B-130, the 3rd paragraph is revised as follows:

      A count of Class 1 welds less than 4 inches nominal pipe size noted approximately ~~400~~ ***450*** welds in the Reactor Coolant (RC), Chemical and Volume Control (CS) and Safety Injection (SI) systems ***within the scope of this program***.~~Approximately 25% of these are socket welds. The number of welds on pipe less than 2 inches nominal pipe size (which includes small branch connections) is less than 50% of the total population.~~ ***The sample***

selection will consist of 10% of the welds within this population and include at least 10% of the total number of socket welds in this population.

Appendix B states that inspection locations will be selected based on susceptibility, inspectability, dose considerations, operating experience, and limiting locations using the recommendations of MRP-146. Appendix B has been clarified to state that the selection will be based on susceptibility, inspectability, dose considerations, operating experience, and limiting locations using MRP-146 evaluations, Probabilistic Risk Assessment risk-ranking and ASME Section XI as guidance. The following change has been made to the Seabrook Station License Renewal Application:

2. In Section B.2.1.23, page B-129, the last paragraph is revised as follows:

   The Seabrook Station ASME Section XI Inservice Inspection, Subsections IWB, IWC and IWD Program, B.2.1.1, currently includes volumetric examination of welds on Class 1 pipe 4 inches nominal pipe size and larger. The Seabrook Station One-Time Inspection of ASME Code Class 1 Small Bore Piping Program will select a sample from the total population of ASME Code Class 1 small bore (less than 4 inches nominal pipe size) piping locations based on susceptibility, inspectability, dose considerations, operating experience, and limiting locations using ***MRP-146 evaluations, Probabilistic Risk Assessment risk-ranking and ASME Section XI as guidance*** ~~the recommendations of MRP-146, "Material Reliability Program: Management of Thermal Fatigue in Normally Stagnant Non-Isolable Reactor Coolant System Branch Lines", or later updated guidance.~~ ***The sample selection will give priority to the locations determined to be most susceptible to stress corrosion cracking and to cyclic loading (including thermal, mechanical and vibrational fatigue).*** The inspection sample determination will include both socket welds and butt welds. If non-destructive volumetric inspection techniques have not been qualified, Seabrook will have the option to remove the weld for destructive examination.

4) As stated in GALL AMP XI.M35 and the Seabrook Station aging management program, sample selection 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. If a specific location is not selected based only on accessibility or radiation exposure, a location with similar susceptibility will be identified to replace it in the sample selection.

5) Socket welds will be examined to the maximum extent possible using methods demonstrated to be able to detect cracking in socket welds. The target volume of the examination will be the socket weld and not, as recommended for MRP-146 examinations, the piping downstream of the socket weld. Appendix B will be clarified to state that the program will inspect for cracking using volumetric examination techniques on selected weld locations. The following change has been made to the Seabrook Station License Renewal Application:

1. In Section B.2.1.23, page B-130, the 5th paragraph is revised as follows:

   The Seabrook Station One-Time Inspection of ASME Code Class 1 Small Bore Piping Program will inspect for cracking in ASME Code Class 1 small-bore piping using volumetric examination techniques ***on selected weld locations*** ~~available~~. Should, evaluation of the inspection results indicate the need for additional examinations, such examinations will be consistent with ASME Section XI, Subsection IWB.

Request for Additional Information (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.

NextEra Energy Seabrook Response:

Seabrook Station will add flash point testing as a requirement for in scope diesel engine lube oil analysis.

Review of industry literature indicates that flash point testing is one traditional technique used to determine if a reduction in flash point temperature has occurred due to the presence of lighter hydrocarbon fuel components, originating from fuel contaminants such as gasoline, diesel and biodiesel. ASTM D 6224-98 "Standard Practice for In-Service Monitoring of Lubricating Oil for Auxiliary Power Plant Equipment", limits its optional test recommendation for flash point testing to diesel engine oils. Discussion in the standard indicates that flash point testing is useful for detecting contamination with diesel fuel or low-boiling solvents; however flash point is of little significance for determining the degree of degradation of used oil since normal degradation has little effect on the flash point.

Therefore, Seabrook Station will limit the flash point testing to those lube oil samples that have the potential for contamination by fuel, which are the two main emergency diesel generators and the two diesel fire pumps.

Since the remainder of the programs lube oil samples do not have a potential for contamination by fuel, no flash point testing will be performed on these samples. Since no flash point testing will be completed on these samples, an exception is still required.

Based on the above discussion, the following changes have been made to the Seabrook Station License Renewal Application:

1. In Section B.2.1.26, on page B-147, revised the 2nd paragraph of the 1st exception to NUREG-1801 as follows:

   ~~Seabrook Station does not sample for flash point in lubricating oil samples.~~ ***Seabrook samples for flash point in the lubricating oil of diesel fueled engines.***

2. In Section B.2.1.26, on page B-147, revised the justification for the 1st exception to NUREG-1801 as follows:

   Testing for flash point of lubricating oil is only needed for lubricating oil that could become contaminated by fuel. Lubricating oil in many applications is not subject to this contamination (such as steam driven turbines or motor driven pumps) and testing for flash point is not needed. When there is no potential for contamination the lube oil will not be tested for flash point. ~~In fact, ASTM D6224, *"Standard Practice for In-Service Monitoring of Lubricating Oil for Auxiliary Power Plant Equipment"*, states that flash point testing is optional for diesel engines.~~

   ~~When there is a potential for contamination by fuel Seabrook Station will test the samples for fuel dilution. This is equivalent to testing for flash point as either test will provide an indication of fuel in-leakage. An example of the ability to detect fuel in-leakage using this method is included in operating experience.~~

   Program Elements Affected: Element 3 (Parameters Monitored/Inspected).

3. In Section B.2.1.26, on pages B-147 and B-148, revised the 1st enhancement as follows:

   The following enhancements will be made prior to entering the period of extended operation.

   The Seabrook Station Lubricating Oil Analysis Program will be enhanced to provide an attachment with required equipment, and include the lube oil analysis required, sampling frequency, and required periodic oil changes. ***The lube oil analysis required will include "Flash Point" when there is a potential for contamination of the lubrication oil by fuel.***

   Program Elements Affected: Element 3 (Parameters Monitored/Inspected) and Element 4 (Detection of Aging Effects)

Request for Additional Information (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.

NextEra Energy Seabrook Response:

Typically, nine BORAL surveillance coupons are shipped to Northeast Technology Corp's (NETCO) Laboratory at the Pennsylvania State University for testing. One archive coupon that had not been in the pool is also furnished for comparative testing. The nine coupons are a subset of the 16 coupons that reside on a special surveillance assembly in the Seabrook Station spent fuel pool.

Prior to shipment to NETCO's laboratory the coupons are removed from their stainless steel capsules by Seabrook Station personnel. After arrival at NETCO's lab, the coupons are subject to visual inspection and high resolution digital photography as specified by NETCO procedures. Following visual inspection, those coupons which had blisters in the cladding are subjected to detailed blister identification and characterization including quantitative determination of blister height and area. This is performed so that the progression of blister growth in subsequent test campaigns can be determined. Additionally, each of the coupons is subjected to neutron radiography and the measurement of boron areal density at specified locations. This work is performed in the Beam Hole Laboratory of the Breazeale Research Reactor at Penn State.

An inspection and testing report prepared by NETCO documents the results of these inspections and tests. All macrophotographs, microphotographs and photomicrographs shown in this report are furnished in electronic form on a compact disk.

The NETCO inspection procedure does provide special precautions and requirements specific to the handling of these coupons. The following are excerpted from that procedure, and represent the level of control put in place by Seabrook Station.

1. All coupon examinations, handling and shipment are to be nondestructive in nature. The coupon physical condition, attributes and appearance shall be unaltered in any way, during examinations and handling that would affect their continued use in the Seabrook Station monitoring program. Coupons shall not be dented, scared, cut or otherwise affected without the expressed written consent of Seabrook Station.

2. Uncontaminated control coupons shall remain uncontaminated.

3. All Seabrook coupon material shall remain the property of Seabrook Station and shall be returned to Seabrook Station within two weeks after the completion of examinations or upon request of Seabrook Station.

4. The unexposed control coupon shall NOT be exposed to water or chemical reagents.

The following excerpts from the Seabrook Station Reactor Engineering Boral Monitoring Program document describe typical precautions and prerequisites in place at the site to ensure consistent and effective handling and testing of the Boral coupons.

1. An examination specification is required prior to engaging contractor coupon inspections services.

2. Care shall be used while handling the coupons to avoid denting, scoring or creasing the 20 mil stainless steel coupon jackets.

3. The coupons should be handled and transported within the numbered stainless steel jackets or placed in numbered plastic bags when removed from their jackets to avoid identification mix-ups. Coupon examinations should be performed one at a time to avoid identification mix-ups.

4. Care should be used in handling the coupons to preserve their appearance and physical condition except as provided for within planned destructive examinations.

5. During examination, noteworthy observations should be made and recorded to note the orientation indicated on the specific coupon photo or diagram. During installation of the coupon into their stainless steel jackets, and installation of the coupons in the train assemblies, the orientation should position the coupon ID number in the upper right hand corner as observed when facing each side of the train in the upright position.

6. All required materials and off-site examination services, including special coupons and neutron attenuation, shall be procured by purchase order to assure design control and quality requirements.

Request for Additional Information (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.

NextEra Energy Seabrook Response:

The following commitment is added to Section A.3 LICENSE RENEWAL COMMITMENT LIST of the License Renewal Application on page A-43:

No.	PROGRAM or TOPIC	COMMITMENT	UFSAR LOCATION	SCHEDULE
59	*Nickel Alloy Nozzles and Penetrations*	***The Nickel Alloy Aging Nozzles and Penetrations program will implement applicable Bulletins, Generic Letters, and staff accepted industry guidelines.***	*A.2.2.3*	***Prior to the period of extended operation.***

Request for Additional Information (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.

NextEra Energy Seabrook Response:

(1) The category (a) fatigue analysis and the category (b) supplemental fatigue analysis were based on transients listed in LRA Table 4.3.1-2.

(2) The Seabrook Station cycle counting procedure tracks and monitors design transients listed in LRA Table 4.3.1-2 but not in TS 5.7 in accordance with the Metal Fatigue of Reactor Coolant Pressure Boundary Program. In the response to RAI B.2.3.1-3, an enhanced Table 4.3.1-2 is provided detailing the current and future proposed monitoring of design transients. The additional design transients shown in the enhanced Table 4.3.1-2 are monitored by the Metal Fatigue of Reactor Coolant Pressure Boundary Program through the Station Engineering Procedure for cycle counting. The FatiguePro automated cycle counting module will be utilized in companion to this procedure to count, categorize and record the plant transients listed in the last column of the table. The additional design transients are selected to record a comprehensive set of plant transients for the purposes of assuring plant operation remains within the fatigue design bases of important plant components and to provide plant operational data should it be necessary to re-analyze plant components in the future. Consistent with License Renewal Commitment #41, Normal, Upset and Test condition transients defined in the Technical Specifications and UFSAR will be monitored during the period of extended operation with the exception of the "Initial and Random Steady state fluctuations and Boron Concentration Equalization" transients. The Steady state fluctuations and Boron Concentration Equalization transients are considered to be insignificant stress events and do not contribute to fatigue usage for any component. Also, since the number of actual transients could not credibly approach the analyzed number of cycles for these transients during the period of extended operation, these transients are therefore not currently counted or proposed to be counted during the Period of Extended Operation. In addition Emergency and Faulted Condition transients listed in Table 4.3.1-2 are not required to be included in fatigue evaluations and therefore are not monitored by the Metal Fatigue of Reactor Coolant Pressure Boundary Program.

Request for Additional Information (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.

NextEra Energy Seabrook Response:

(a.) Several wording differences are noted between the transient designations listed in LRA Table 4.3.1-2 and those defined in the UFSAR Section 3.9(N).1.1 and the Metal Fatigue of Reactor Coolant Pressure Boundary Program Basis Document. As shown the attached enhanced Table 4.3.1-2 in the response to RAI B.2.3.1-3, the differences fall into three categories:

1. Differences in the use of abbreviations and combining of symmetric transients (e.g., two separate transients listed as Unit Loading and Unloading at 5 Percent of Full Power per Minute compared to two transients individually listed as Unit Loading @ 5% full power/min and Unit Unloading @ 5% full power/min).

2. Addition of supplementary auxiliary transients necessary for the fatigue design basis for the CVCS components.

3. Several fatigue-insignificant plant transients are omitted from the PBD (e.g., Steady-State Fluctuations and Boron Concentration Equalization).

The transients listed in the first column of Table 4.3.1-2 are included in the Fatigue Management Program. Seabrook Station procedure for Documentation of Design Operating Transients is used to identify, count and categorize those plant transients. The fatigue analyses of the Class 1 components are based on the set of plant transients listed in Table 4.3.1-2. These design fatigue analyses are limited to the design limit of 1.0. The Fatigue Management Program will ensure that all fatigue-analyzed components remain within their respective design fatigue analyses results by assuring that the counted plant transients remain within the number of occurrences of each plant transient listed in the second column of Table 4.3.1-2. In this way, the Fatigue Management Program will assure that all fatigue analyses remain valid.

(b.) The six auxiliary transients listed in the enhanced Table 4.3.1-2 with Footnote (6) are specified in the design specification for defining fatigue transients for CVCS components. The fatigue analyses of the Class 1 CVCS components include the normal, upset and test transients in the UFSAR supplemented by the six auxiliary transients. These auxiliary transients are identified and counted in the FatiguePro software that is part of the Fatigue Management Program.

Request for Additional Information (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.

NextEra Energy Seabrook Response:

(a.) An enhanced LRA Table 4.3.1-2 is attached which shows the additional design transients which will be monitored by the Metal Fatigue of Reactor Coolant Pressure Boundary Program through the Seabrook Station procedure for Documentation of Design Operating Transients. The FatiguePro automated cycle counting module will be utilized as a companion to this procedure to count, categorize and record the plant transients listed in the last column of the table. The additional design transients are selected to record a comprehensive set of plant transients for the purposes of assuring plant operation remains within the fatigue design bases of important plant components and to provide plant operational data should it be necessary to re-analyze plant components in the future.

(b.) The additional transients are postulated in the design basis calculations, but were not included in the TS 5.7 and UFSAR Section 3.9.

(c.) The additional transients were included in the environmentally-assisted fatigue evaluations that were prepared for license renewal in LRA 4.3.4

Based on the above discussion, the following change has been made to the Seabrook Station License Renewal Application:

1. In LRA Section 4.3.1, (pages 4.3-4, 4.3-5) an enhanced Table 4.3.1-2 is provided to identify transients which will be monitored by the Metal Fatigue of Reactor Coolant Pressure Boundary Program through the Seabrook Station procedure for Documentation of Design Operating Transients. The FatiguePro automated cycle counting module will be utilized as a companion to this procedure to count, categorize and record the plant transients listed in the last column of the table. The additional design transients are selected to record a comprehensive set of plant transients for the purposes of assuring plant operation remains within the fatigue design bases of important plant components and to provide plant operational data should it be necessary to re-analyze plant components in the future.

Table 4.3.1-2 Summary of Reactor Coolant System Design Transients					
Transient Description	Limiting Design Basis Number of Occurrences for 40 Year Operating Period	UFSAR Section 3.9(N).1.1	Aging Management Program Basis Document Metal Fatigue of Reactor Coolant Pressure Boundary	Counted in	
				Station Procedure	FatiguePro
Normal Condition Transients:					
Plant Heatup @ ≤ 100 °F/hr	200	Heatup and Cooldown at 100°F per hour	Plant (RCS) Heatup	Y	Y
Plant Cooldown @ ≤ 100 °F/hr	200		Plant (RCS) Cooldown	Y	Y
Pressurizer Heatup	200	Not Specified		N	Y
Pressurizer Cooldown	200	Pressurizer cooldown 200°F per hour		N	Y
Unit Loading @ 5% full power/min	13,200 (1)	Unit Loading and Unloading at 5 Percent of Full Power per Minute		Y	Y
Unit Unloading @ 5% full power/min	13,200 (1)			Y	Y
Step Load Increase of 10% of full power	2,000	Step Load Increase and Decrease of 10 Percent of Full Power		Y	Y
Step Load Decrease of 10% of full power	2,000			Y	Y
Large step load decrease with steam dump	200	Large Step Load Decrease with Steam Dump	Large Step Load Decrease	Y	Y
Steady state fluctuations (7)	Initial – 1.5 x 10^5 Random – 3.0 x 10^5	Steady-State Fluctuations		N	N
Feedwater Cycling at Hot Shutdown	2,000	Feedwater Cycling at Hot Shutdown	Feedwater Cycling	N	Y

Table 4.3.1-2 Summary of Reactor Coolant System Design Transients					
Transient Description	Limiting Design Basis Number of Occurrences for 40 Year Operating Period	UFSAR Section 3.9(N).1.1	Aging Management Program Basis Document Metal Fatigue of Reactor Coolant Pressure Boundary	Counted in	
				Station Procedure	FatiguePro
Loop out of service Normal loop shutdown Normal loop startup	 80 70	Loop out of service (4) Normal loop shutdown Normal loop startup		N Y	Y Y
Feedwater Heaters out of service One heater out of service One bank of heaters out of service	 120 120	Feedwater Heaters out of service One heater out of service One bank of heaters out of service		N	Y
Unit loading between 0% to 15% of full power	500 (2)	Unit Loading and Unloading Between 0 and 15 Percent of Full Power		Y	Y
Unit unloading between 15% to 0% of full power	500 (2)			Y	Y
Boron concentration equalization (8)	26,400	Boron Concentration Equalization		N	N
Refueling	80	Refueling	Refueling	Y	Y
Reduced temperature return to power	2,000	Reduced Temperature Return to Power		Y	Y
Reactor Coolant Pumps startup/shutdown	3,000 (3)	Reactor Coolant Pumps (RCP) Startup and Shutdown		Y	Y

Table 4.3.1-2 Summary of Reactor Coolant System Design Transients					
Transient Description	Limiting Design Basis Number of Occurrences for 40 Year Operating Period	UFSAR Section 3.9(N).1.1	Aging Management Program Basis Document Metal Fatigue of Reactor Coolant Pressure Boundary	Counted in	
				Station Procedure	FatiguePro
Letdown Flow Step Decrease and Return (6)	2,000	Not Specified	Letdown Flow Step Decrease and Return	N	Y
Upset Transients:					
Loss of load without immediate turbine trip	80	Loss of Load (Without Immediate Turbine Trip)	Loss of Turbine Load	Y	Y
Loss of all offsite power (blackout with natural circulation in the RCS)	40	Loss of Power	Loss of Offsite Power	Y	Y
Partial loss of flow (loss of one pump)	80	Partial Loss of Flow	Partial Loss of RCS Flow	Y	Y
Reactor trip from full power: *Without cooldown* *With cooldown, without safety injection* *With cooldown and safety injection*	 230 160 10	Reactor Trip from Full Power: *Reactor trip with no inadvertent cooldown* *Reactor trip with cooldown but no safety injection* *Reactor trip with cooldown actuating safety injection*	-Reactor Trip from Full Power – with no Inadvertent Cooldown -Reactor Trip from Full Power – with Cooldown and no SI -Reactor Trip from Full Power – with Cooldown and SI (HHSI)	Y Y Y	Y Y Y

Table 4.3.1-2 Summary of Reactor Coolant System Design Transients					
Transient Description	Limiting Design Basis Number of Occurrences for 40 Year Operating Period	UFSAR Section 3.9(N).1.1	Aging Management Program Basis Document Metal Fatigue of Reactor Coolant Pressure Boundary	Counted in	
				Station Procedure	FatiguePro
Inadvertent reactor coolant depressurization	20	Inadvertent Reactor Coolant System Depressurization	Inadvertent RCS Depressurization	Y	Y
			Inadvertent Pressurizer Auxiliary Spray Actuation (5)	Y	Y
Inadvertent startup of inactive loop	10	Inadvertent Startup of an Inactive Loop		Y	Y
Control rod drop	80	Control Rod Drop		Y	Y
Inadvertent ECCS actuation	60	Inadvertent Safety Injection Actuation	Inadvertent Safety Injection (SI) Actuation	Y	Y
Operating Basis Earthquake (5 earthquakes of 10 cycles each)	50	Operating Basis Earthquake	Operating Basis Earthquake (OBE)	N	Y
Excessive feedwater flow	30	Excessive Feedwater Flow	Excessive Feedwater Flow	Y	Y
RCS Cold Overpressurization	10	RCS Cold Overpressurization		N	Y
Charging and Letdown Flow Shutoff and Return (6)	60	Not Specified	Charging and Letdown Flow Shutoff and Return	N	Y
Charging Flow Shutoff with Delayed Return (6)	20	Not Specified	Charging Flow Shutoff with Delayed Return	N	Y
Charging Flow Shutoff with Prompt Return (6)	20	Not Specified	Charging Flow Shutoff with Prompt Return	N	Y
Letdown Flow Shutoff with Delayed Return (6)	20	Not Specified	Letdown Flow Shutoff with Delayed Return	N	Y

Table 4.3.1-2 Summary of Reactor Coolant System Design Transients					
Transient Description	Limiting Design Basis Number of Occurrences for 40 Year Operating Period	UFSAR Section 3.9(N).1.1	Aging Management Program Basis Document Metal Fatigue of Reactor Coolant Pressure Boundary	Counted in	
				Station Procedure	FatiguePro
Letdown Flow Shutoff with Prompt Return (6)	200	Not Specified	Letdown Flow Shutoff with Prompt Return	N	Y
Emergency Transients:					
Small LOCA	5	Small Loss-of-Coolant Accident		N	N
Small steam break	5	Small Steam Line Break		N	N
Complete loss of flow	5	Complete Loss of Flow		N	N
Faulted Transients:					
Main reactor coolant pipe break (LOCA)	1	Reactor Coolant Pipe Break (Large Loss-of-Coolant Accident)		N	N
Large steam line break	1	Large Steam Line Break		N	N
Feedwater line break	1	Feedwater Line Break		N	N
Reactor Coolant Pump locked rotor	1	Reactor Coolant Pump Locked Rotor		N	N
Control rod ejection	1	Control Rod Ejection		N	N
Steam Generator tube rupture	Included under Reactor Trip with cooldown and safety injection	Steam Generator Tube Rupture		N	N

Table 4.3.1-2 Summary of Reactor Coolant System Design Transients					
Transient Description	Limiting Design Basis Number of Occurrences for 40 Year Operating Period	UFSAR Section 3.9(N).1.1	Aging Management Program Basis Document Metal Fatigue of Reactor Coolant Pressure Boundary	Counted in	
				Station Procedure	FatiguePro
Safe Shutdown Earthquake	1	Safe Shutdown Earthquake		N	N
Test Transients:					
Primary side hydrostatic test	10	Primary Side Hydrostatic test	Primary Side RCS Hydrostatic Test	Y	Y
Secondary side hydrostatic test	10	Secondary Side Hydrostatic Test		Y	Y
Turbine roll test	20	Turbine Roll Test	Turbine Roll Test	Y	Y
Primary side leak test	200	Primary Side Leakage Test	Primary Side RCS Leakage Test	Y	Y
Secondary side leak test	80	Secondary Side Leakage Test		Y	Y
Tube leak test	800	Tube Leakage Test		Y	Y

1. For the design transient of Unit Loading and Unit Unloading @ 5% full power/min., the Reactor Vessel, Steam Generators and Pressurizers are designed for 13,200 cycles, where the Class 1 piping is designed for 18,300 cycles. The most limiting value of these major components is used as a monitoring limit in the Metal Fatigue of Reactor Coolant Pressure Boundary Program (B.2.3.1).

2. For the design transients of Unit load and unload between 0% to 15% of full power, the Reactor Vessel, Steam Generators and Class 1 piping are designed for 500 cycles, where the Pressurizer is designed for 1,510 cycles. The most limiting value of these major components is used as a monitoring limit in the Metal Fatigue of Reactor Coolant Pressure Boundary Program (B.2.3.1).

3. For the design transient of Reactor Coolant Pump startup/shutdown, the limit specified in the UFSAR is 3800 cycles. The Pressurizer is designed for 4,000 cycles, where Steam Generators are designed for 3,000 cycles. The Steam Generators has the

most limiting value (3,000 cycles) of these components is used as a monitoring limit in the Metal Fatigue of Reactor Coolant Pressure Boundary Program (B.2.3.1).

4. Categorization of the Loop out of Service transient is taken from UFSAR Section 3.9(N).1.1.a.7.
5. Inadvertent Pressurizer Auxiliary Spray Actuation transient specified in the PBD is one of the five subevents included in the Inadvertent Reactor Coolant System Depressurization event.
6. Transients identified as auxiliary transients in Westinghouse Systems Standard.
7. The Steady-State Fluctuation event does not contribute to the computed fatigue usage for any analyzed component and is not specifically counted.
8. The Boron Concentration Equalization event is a load-following event, and is not specifically counted,

Request for Additional Information (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.

NextEra Energy Seabrook Response:

An action limit of 80% will be used by the Metal Fatigue of Reactor Coolant Pressure Boundary Program for all categories as described below. This action limit will provide sufficient margin and time to allow for appropriate corrective actions as defined in the Metal Fatigue of Reactor Coolant Pressure Boundary Program to be implemented prior to reaching the design limit.

The action limit for components for which CUF values are evaluated against a design limit of CUF = 1.0 is 80% of the design or analyzed number of occurrences for any plant transient included in the fatigue analysis of any of these fatigue-analyzed components. When any one or more of these plant transients reaches a value of 80% or more of the design number of occurrences, a re-evaluation of each of the components analyzed for the plant transient(s) that have achieved their 80% limit will be performed. This re-evaluation may take advantage of use of 60-year projected cycles of other plant transients that participate in the fatigue analysis of each of the affected components.

The action limit for components for which environmentally-assisted CUF values are evaluated against a design limit of CUF = 1.0 is 80% of the analyzed number of occurrences for any plant transient included in the fatigue analysis of any of these EAF-analyzed components. When any one or more of these plant transients reaches a value of 80% or more of the analyzed number of occurrences, a re-evaluation of each of the components analyzed for the plant transient(s) that have achieved their 80% limit will be performed. This re-evaluation may take advantage of use of 60-year projected cycles of other plant transients that participate in the fatigue analysis of each of the affected component.

The action limit for HELB components for which CUF values are evaluated against a design limit of CUF = 0.1 is 80% of the analyzed number of occurrences for any plant transient included in the fatigue analysis of any of these EAF-analyzed components. When any one or more of these plant transients reaches a value of 80% or more of the analyzed number of occurrences, a re-evaluation of each of the components analyzed for the plant transient(s) that have achieved their 80% limit will be performed. This re-evaluation may take advantage of use of 60-year projected cycles of other plant transients that participate in the fatigue analysis of each of the affected component.

Request for Additional Information (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

NextEra Energy Seabrook Response:

GALL AMP X.M1 program element does not specifically address the use of a fracture mechanics evaluation to reanalyze affected components. However, the ASME Boiler and Pressure Vessel Code, Section XI, Appendix L, provides guidance for methods for performing fatigue assessments to determine acceptability for continued service of reactor coolant system and primary pressure boundary components subjected to thermal and mechanical fatigue loads.

The AMSE Code specifies two methods for performing fatigue assessments; a fatigue usage factor evaluation and a flaw tolerance evaluation (using fracture mechanics techniques). These two evaluation methods are the analytical options provided in LRA Section B.2.3.1. The use of the flaw tolerance evaluation method would require NRC approval of the fatigue crack growth curves used in the evaluation.

Request for Additional Information (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.

NextEra Energy Seabrook Response:

(a.) The EPRI FatiguePro program is used to perform several functions to accommodate fatigue monitoring of fatigue-critical components. The data acquisition system module collects plant instrument data for selected time periods. The automated cycle counting module analyzes the collected plant instrument data and identifies, counts, categorizes and records pre-defined plant transients with their pertinent engineering parameters. The cycle-based fatigue module calculates cumulative fatigue usage for selected plant components by applying counted plant transients to the component design stress report fatigue analysis.

The FatiguePro software will be used in conjunction with the Seabrook Station cycle counting procedure to provide the technical basis and data for monitoring the number of occurrences and severity of the plant transients that define the fatigue design basis for fatigue-critical components. This data will be used to assure that the number and severity of the counted plant transients remain bounded by the component design analyses. In this way, assurance can be provided that the design fatigue usage limit of 1.0 is maintained.

(b.) As described in (a) above, the EPRI FatiguePro cycle-based fatigue program calculates cumulative fatigue usage for selected plant components by applying counted plant transients to the component design stress report fatigue analysis. This cumulative fatigue usage computation is used as a secondary method for detecting aging effects. The primary method for detecting aging effects is the cycle counting program. This method is used to assure that the number and severity of the counted plant transients remain bounded by the component design analyses. In this way, assurance can be provided that the design fatigue usage limit of 1.0 is maintained.

(c.) NRC RIS 2008-30 addresses the use of simplifying assumptions when performing new ASME Section III NB-3200 fatigue analyses. The concern is that simplification of the six-stress tensor state to fewer stress components in the course of the fatigue analysis may lead to un-conservative fatigue usage results and should be benchmarked to six-stress tensor analyses whenever used. This simplified method has been utilized in the Stress-Based Fatigue module of the current FatiguePro software and is only utilized in that module. To resolve and address this issue the Seabrook Station Metal Fatigue of Reactor Coolant Pressure Boundary Program will not utilize the simplified FatiguePro Stress-Based Fatigue module.

Request for Additional Information (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 5a) 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 5b) 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 5c) provide additional calculations which, considering the potential for PWSCC, demonstrate that the principles of LBB analyses are satisfied.

NextEra Energy Seabrook Response:

1) Leak Before Break (LBB) is applicable only to the Reactor Coolant Loop piping. Materials susceptible to PWSCC (Alloys 600/82/182) were utilized only in the Reactor Vessel inlet and outlet nozzle safe-end welds.

2) The Seabrook Station inspection program for materials susceptible to PWSCC (Alloys 600/82/182) is documented in the Seabrook Station RCS Materials Degradation Management Reference Manual. The program adheres to the guidelines documented in EPRI MRP-139

Revision 1, "Materials Reliability Program: Primary System Piping Butt Weld Inspection and Evaluation Guidelines".

During refueling outage OR013 in the fall of 2009, all eight reactor vessel nozzle butt welds were volumetrically inspected. This met the mandatory inspection deadline of December 31, 2009 for the hot legs and December 31, 2010 for the cold legs. During the inspection an axial flaw indication was found on the reactor vessel loop 158 degree hot leg nozzle in the alloy 82/182 material connected to the inner diameter. That nozzle was subsequently mitigated by MSIP. MRP-139 has a mandatory examination requirement for ultrasonic testing (UT) for post stress improvement mitigation. Due to the hardship associated with disassembly of the reactor components to perform the examination, Seabrook Station decided to delay the post MSIP examination until OR014, Spring 2011. This deviation from the MRP-139 requirement was documented in Seabrook Engineering Evaluation EE-09-017. Since the implementation of MRP-139 is a "Mandatory" requirement under the NEI 03-08 Material Initiative this deviation and its technical justification were prepared in accordance with the NEI 03-08 protocol and approved by the FPL Chief Nuclear Officer. Seabrook Station will perform a baseline UT examination of the 158^{o} hot leg nozzle to safe end dissimilar metal weld in OR014 Spring 2011 and perform subsequent inspections to come into full compliance with MRP-139. Other alloy 600 locations inspected in OR013 were the reactor vessel bottom mounted nozzles, reactor vessel o-ring leak off lines, and the steam generator bowl drains. These inspections were satisfactory.

The Pressurizer Components, Control Rod Drive Pressure Housing, Reactor Vessel Bottom Instrument Tube, Steam Generator Channel Head Drain Pipe, and the Steam Generator Primary Nozzle Weld are age managed by the ASME Section XI Inservice Inspection Subsections IWB IWC and IWD Program, Water Chemistry Program, and Nickel Alloy Nozzles and Penetrations Program.

The Reactor Vessel Head Vent Pipe, Reactor Vessel Control Rod Drive Penetration Nozzle and Welds are age managed by the ASME Section XI Inservice Inspection Subsections IWB IWC and IWD Program, Water Chemistry Program, and Nickel Alloy Penetration Nozzles Welded to the Upper RV Closure Heads of PWRs Program.

The Reactor Vessel Primary Inlet and Outlet Nozzle Welds are age managed by the ASME Section XI Inservice Inspection Subsections IWB IWC and IWD Program and Water Chemistry Program.

3) During the October 2009 Cycle 13 Refueling Outage; Seabrook Station was performing the second 10 year UT in-service inspection of its reactor vessel. During the inspection, performed by WesDyne International, a subsidiary of Westinghouse Electric, flaw indications were identified in 2 of 4 of the vessel outlet (hot leg) nozzle to safe end welds.

As documented on the WesDyne Indication Data Sheet # SIZ-DM-22-1 for the outlet nozzle @ 22 degrees, two indications were detected. WesDyne concluded the following:

"Indications 1 & 2 are wholly contained within the clad thus no IWB-3500 evaluations are required Per IWB 3514.1(d)(1). Similar indications, but below recordable levels, were also

seen in the same location randomly around the entire weld."

The WesDyne Indication Data Sheet # SIZ-DM-158-1 for the outlet nozzle @158 degrees documents that two indications were detected. For Indication 2, WesDyne concluded the following:

"Indication 2 is wholly contained within the clad thus no IWB-3500 evaluations is required per IWB 3514.1(d)(1). Similar indications to indication #2, but below recordable levels, were also seen in the same location randomly around the entire weld."

WesDyne prepared an Indication Assessment Sheet # DM-158-1 for Indication #1. The indication characteristics are shown in the table below.

NOZZLE LOCATION	LOCATION DESCRIPTION	FLAW DIRECTION	FLAW DEPTH/THICKNESS RATIO A/T (%)	FLAW SHAPE A/L
Outlet Nozzle @ 158 degrees	ID @ 93 degrees	Axial Surf	a =0.614; t = 2.925 a/t = 0.2099 20.99%	a = 0.614; l = 0.96 a/l = 0.6396 (0.50 max value)

As noted on the assessment sheet, a correction factor of 0.064 inch was added to the through wall depth. The assessment sheet concluded that the Indication #1 results for the 158 degree nozzle were unacceptable per the ASME Section XI Code acceptance standards.

4) A Mechanical Stress Improvement Process (MSIP) repair was performed on the Reactor Vessel 158 degree outlet nozzle in Refueling Outage 13. Engineering Change (EC) 145179 authorized a nozzle repair by MSIP of the Seabrook Station Reactor Vessel hot leg nozzle and safe end.

   The implementation of MSIP provides a mitigation of PWSCC of reactor pressure vessel nozzles (hot and cold legs) to safe-end welds by applying a narrow permanent radial deformation on a small section of the pipe so a beneficial stress profile is created through the cross-section of the weld region. The stress profile mitigates PWSCC by setting up compressive stresses in place of tensile residual stresses.

5a) See response to item 5b)

5b) The applied MSIP discussed in response 4) above was performed by Westinghouse. As part of the application process, Westinghouse reviewed the impact on the Leak-Before-Break Evaluation for Seabrook Station documented in WCAP-10567, "Technical Bases for Eliminating Large Primary Loop Pipe Rupture as the Structural Design Basis for Seabrook Units 1 and 2, June 1984." In their letter, LTR-PAFM-09-131, October 29, 2009, Westinghouse concluded that the compressive residual stresses generated at the butt welds due to the MSIP application will have a favorable impact on the Fatigue Crack Growth (FCG) analysis, such that a part through-wall crack will not be able to become a complete through-wall axial crack. Furthermore, Westinghouse concluded that the axial flaw found will not be critical for the LBB analysis, as it is restricted to the width of the butt welds, and the primary loading is due to pressure only. Therefore, the LBB conclusions shown in WCAP-10567 for Seabrook Station remain valid for the MSIP application.

5c) See response to item 5b)

Request for Additional Information (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.

NextEra Energy Seabrook Response:

The original plant design leak before break calculation does constitute a TLAA as described in LRA section 4.7.3.

Based on the above discussion, the following change has been made to the Seabrook Station License Renewal Application:

1. An updated analysis and conclusion section for LRA Section 4.7.13 (page 4.7-14) is provided

below removing the second bullet to clarify that the original plant design leak before break calculations does constitute a TLAA as described in LRA section 4.7.3.

Analysis

A thorough review of the Seabrook Station licensing basis, supported by interviews with plant staff familiar with the history of Class 1 components, found the following fatigue crack growth analyses:

- Fatigue crack growth and fracture mechanics stability analyses in support of pressurizer nozzle overlays. The overlays were supported by fatigue crack growth analyses. These fatigue crack growth analyses were projected to the end of the period of extended operation, and are therefore not TLAAs. See Section 4.3.6.

- ~~Fatigue crack growth assessments and fracture mechanics stability analyses in support of the leak-before-break (LBB) evaluation. Review for the period of extended operation was based on the same set of design transients as the 40-year analyses; therefore the conclusions of the original evaluation remain valid for the 60-year period and are therefore not TLAAs. See Section 4.7.3.~~

- Fatigue crack growth and fracture mechanics stability analyses of Mechanical Stress Improvement Process (MSIP) repairs to Alloy 600 material in reactor coolant hot legs. The MSIP repair was supported by fatigue crack growth analysis. This fatigue crack growth analysis was projected to the end of the period of extended operation, and is therefore not a TLAA. See Section 4.3.6.

Conclusion

These fatigue crack growth analyses are not TLAAs because they qualify the affected components for the period of extended operation. ***Fatigue crack growth assessments and fracture mechanics stability analyses in support of the leak-before-break (LBB) evaluation was reviewed for the period of extended operation and was based on the same set of design transients as the 40-year analyses; therefore the conclusions of the original evaluation remain valid for the 60-year period as discussed in Section 4.7.3.***