NEXTera ENERGY SEABROOK

December 12, 2013

Docket No. 50-443
SBK-L-13225

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

Seabrook Station
Submittal of Changes to the Seabrook Station Technical Specification Bases

NextEra Energy Seabrook, LLC submits the enclosed changes to the Seabrook Station Technical Specification Bases. The changes were made in accordance with Technical Specification 6.7.6.j., "Technical Specification (TS) Bases Control Program." Please update the Technical Specification Bases with the enclosed pages as follows:

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Should you have any questions concerning this submittal, please contact me at (603) 773-7512.

Sincerely,

NextEra Energy Seabrook, LLC

Michael H. Ossing
Licensing Manager

cc: NRC Region I Administrator
NRC Project Manager, Project Directorate I-2
NRC Senior Resident Inspector

Enclosure to SBK-L-13225

3/4.4.5 STEAM GENERATORS (SG) TUBE INTEGRITY (Continued)

APPLICABILITY

Steam generator tube integrity is challenged when the pressure differential across the tubes is large. Large differential pressures across SG tubes can only be experienced in MODES 1, 2, 3, or 4.

RCS conditions are far less challenging in MODES 5 and 6 than during MODES 1, 2, 3, and 4. In MODES 5 and 6, primary to secondary differential pressure is low, resulting in lower stresses and reduced potential for leakage.

ACTIONS

The ACTIONS are modified by a Note clarifying that the actions may be entered independently for each SG tube. This is acceptable because the actions provide appropriate compensatory actions for each affected SG tube. Complying with the actions may allow for continued operation, and subsequent affected SG tubes are governed by subsequent entry and application of associated actions.

a and b

Action a applies if it is discovered that one or more SG tubes examined in an inservice inspection satisfy the tube plugging criteria but were not plugged in accordance with the Steam Generator Program as required by SR 4.4.5.2. An evaluation of SG tube integrity of the affected tube(s) must be made. Steam generator tube integrity is based on meeting the SG performance criteria described in the Steam Generator Program. The SG plugging criteria define limits on SG tube degradation that allow for flaw growth between inspections while still providing assurance that the SG performance criteria will continue to be met. In order to determine if a SG tube that should have been plugged has tube integrity, an evaluation must be completed that demonstrates that the SG performance criteria will continue to be met until the next refueling outage or SG tube inspection. The tube integrity determination is based on the estimated condition of the tube at the time the situation is discovered and the estimated growth of the degradation prior to the next SG tube inspection. If it is determined that tube integrity is not being maintained, Action b applies.

A completion time of 7 days is sufficient to complete the evaluation while minimizing the risk of plant operation with a SG tube that may not have tube integrity.

If the evaluation determines that the affected tube(s) have tube integrity, Action a allows plant operation to continue until the next refueling outage or SG inspection provided the inspection interval continues to be supported by an operational assessment that reflects the affected tubes. However, the affected tube(s) must be plugged prior to entering MODE 4 following the next refueling outage or SG inspection. This completion time is acceptable since operation until the next inspection is supported by the operational assessment.

3/4.4.5 STEAM GENERATORS (SG) TUBE INTEGRITY (Continued)

If SG tube integrity is not being maintained, the reactor must be brought to MODE 3 within 6 hours and MODE 5 within 36 hours. The shutdown times are reasonable, based on operating experience, to reach the desired plant conditions from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

4.4.5.1

During shutdown periods, the SGs are inspected as required by this SR and the Steam Generator Program. NEI 97-06, Steam Generator Program Guidelines (Ref. 1), and its referenced EPRI Guidelines, establish the content of the Steam Generator Program. Use of the Steam Generator Program ensures that the inspection is appropriate and consistent with accepted industry practices.

During SG inspections, a condition monitoring assessment of the SG tubes is performed. The condition monitoring assessment determines the "as found" condition of the SG tubes. The purpose of the condition monitoring assessment is to ensure that the SG performance criteria have been met for the previous operating period.

The Steam Generator Program determines the scope of the inspection and the methods used to determine whether the tubes contain flaws satisfying the tube plugging criteria. Inspection scope (i.e., which tubes or areas of tubing within the SG are to be inspected) is a function of existing and potential degradation locations. The Steam Generator Program also specifies the inspection methods to be used to find potential degradation. Inspection methods are a function of degradation morphology, nondestructive examination (NDE) technique capabilities, and inspection locations. The portion of the SG tubes below 15.21 inches from the top of the tubesheet is excluded from periodic inspections and plugging.

The Steam Generator Program defines the Frequency of SR 4.4.5.1. The Frequency is determined by the operational assessment and other limits in the SG examination guidelines (Ref. 6). The Steam Generator Program uses information on existing degradations and growth rates to determine an inspection frequency that provides reasonable assurance that the tubing will meet the SG performance criteria at the next scheduled inspection. In addition, Specification 6.7.6.k contains prescriptive requirements concerning inspection intervals to provide added assurance that the SG performance criteria will be met between scheduled inspections. If crack indications are found in any SG tube, the maximum inspection interval for all affected and potentially affected SGs is restricted by Specification 6.7.6.k until subsequent inspections support extending the inspection interval.

3/4.4.5 STEAM GENERATORS (SG) TUBE INTEGRITY (Continued)

SR 4.4.5.2

During a SG inspection, any inspected tube that satisfies the Steam Generator Program plugging criteria is removed from service by plugging. The tube plugging criteria delineated in Specification 6.7.6.k are intended to ensure that tubes accepted for continued service satisfy the SG performance criteria with allowance for error in the flaw size measurement and for future flaw growth. In addition, the tube plugging criteria, in conjunction with other elements of the Steam Generator Program, ensure that the SG performance criteria will continue to be met until the next inspection of the subject tube(s). Reference 1 provides guidance for performing operational assessments to verify that the tubes remaining in service will continue to meet the SG performance criteria.

The frequency of prior to entering MODE 4 following a SG inspection ensures that the Surveillance has been completed and all tubes meeting the plugging criteria are plugged prior to subjecting the SG tubes to significant primary to secondary pressure differential.

REFERENCES

1. NEI 97-06, "Steam Generator Program Guidelines."

2. 10 CFR 50 Appendix A, GDC 19.

3. 10 CFR 50.67

4. ASME Boiler and Pressure Vessel Code, Section III, Subsection NB.

5. Draft Regulatory Guide 1.121, "Basis for Plugging Degraded Steam Generator Tubes," August 1976.

6. EPRI, "Pressurized Water Reactor Steam Generator Examination Guidelines."

3/4.4.9 PRESSURE/TEMPERATURE LIMITS (Continued)

The P/T limits have been established in accordance with the requirements of ASME Boiler and Pressure Vessel Code Section XI, Appendix G, as modified by ASME Code Case N-641, Reference (2), and the additional requirements of 10CFR50 Appendix G, Reference (3). The heatup and cooldown P/T limit curves for normal operation, Figures 3.4-2 and 3.4-3 respectively, are valid for a service period of 23.7 effective full power years (EFPY). The technical justification and methodologies utilized in their development are documented in WCAP-15745, Reference (4), and LTR-AMLRS-11-50, Reference (7). The P/T curves were generated based on the latest available reactor vessel information and latest calculated fluences.

Heatup and Cooldown limit curves are calculated using the adjusted RT_{NDT} (reference nil-ductility temperature) corresponding to the limiting beltline region material of the reactor vessel. The adjusted RT_{NDT} of the limiting material in the core region of the reactor vessel is determined by using the unirradiated reactor vessel material fracture toughness properties, estimating the radiation-induced ΔRT_{NDT}, and adding a margin. RT_{NDT} increases as the material is exposed to fast-neutron radiation. Therefore, to find the most limiting RT_{NDT} at any time period in the reactor's life, ΔRT_{NDT} due to the radiation exposure associated with that time period must be added to the unirradiated RT_{NDT} (IRT_{NDT}). The extent of the shift in RT_{NDT} is enhanced by certain chemical elements (such as copper and nickel) present in reactor vessel steels. The Nuclear Regulatory Commission (NRC) has published a method for predicting radiation embrittlement in Regulatory Guide 1.99, Revision 2, Reference (5). Regulatory Guide 1.99, Revision 2, is used for the calculation of Adjusted Reference Temperature (ART) values (IRT_{NDT} + ΔRT_{NDT} + margins for uncertainties) at the 1/4T and 3/4T locations, where T is the thickness of the vessel at the beltline region.

The reactor vessel materials have been tested to determine their initial RT_{NDT}. Reactor operation and resultant fast neutron (E greater than 1 MeV) irradiation can cause an increase in the RT_{NDT}. Therefore, an adjusted reference temperature, based upon the fluence, best estimate copper and nickel content of the limiting beltline material, can be predicted using surveillance capsule data and the value of ΔRT_{NDT} computed by Regulatory Guide 1.99, Revision 2. Surveillance capsule data, documented in Reference (6), is available for two capsules (Capsules U and Y) having already been removed from the reactor vessel. This surveillance capsule data was used to calculate chemistry factor (CF) values per Position 2.1 of Regulatory Guide 1.99, Revision 2. It also noted that Reference (6) concluded that all the surveillance data was credible as the beltline material was behaving as empirically predicted. The heatup and cooldown limit curves of Figures 3.4-2 and 3.4-3 include predicted adjustments for this shift in RT_{NDT} as well as adjustments for possible errors in the pressure and temperature sensing instruments.

3/4.4.9 PRESSURE/TEMPERATURE LIMITS (Continued)

HEATUP (Continued)

10 CFR Part 50, Appendix G, Reference (3), addresses the metal temperature of the closure head flange and vessel flange regions. This rule states that the metal temperature of the closure flange regions must exceed the material unirradiated RT_{NDT} by at least 120°F for normal operation when the pressure exceeds 20 percent of the preservice hydrostatic test pressure (3106 psi), which in this case is 621 psig. The limiting unirradiated RT_{NDT} of 30°F occurs in the vessel flange of the reactor vessel, consequently the minimum allowable temperature of this region is 150°F at pressures greater than 621 psig. This limit is shown as the horizontal lines in Figures 3.4-2 and 3.4-3. (NOTE: Figures 3.4-2 and 3.4-3 include a compensation of 20°F and 100 psig for possible instrument errors.)

Although the pressurizer operates in temperature ranges above those for which there is reason for concern of nonductile failure, operating limits are provided to assure compatibility of operation with the fatigue analysis performed in accordance with the ASME Code requirements.

References

1. ASME Boiler and Pressure Vessel Code, Section XI, Appendix G, "Fracture Toughness Criteria for Protection Against Failure", dated December 1995, through 1996 Addendum.

2. ASME Boiler and Pressure Vessel Code Case N-641, Section XI, Division 1, "Alternative Pressure-Temperature Relationship and Overpressure Protection System Requirements", dated January 17, 2000.

3. 10 CFR Part 50, Appendix G, "Fracture Toughness Requirements", U.S. Nuclear Regulatory Commission, Federal Register, Volume 60, No. 243, dated December 19, 1995.

4. Westinghouse WCAP-15745, Revision 0, "Seabrook Unit 1 Heatup and Cooldown Limit Curves for Normal Operation", dated December 2001.

5. Regulatory Guide 1.99, Revision 2, "Radiation Embrittlement of Reactor Vessel Materials", U. S. Nuclear Regulatory Commission, dated May 1988.

6. Duke Engineering and Services Report DES-NFQA-98-01, Revision 0, "Analysis of Seabrook Station Unit I Reactor Vessel Surveillance Capsules U and Y", dated May 1998.

7. Westinghouse Letter LTR-AMLRS-11-50, Rev. 0, "Seabrook Unit 1 Heatup and Cooldown Limit Curves Applicability Evaluation", August 3, 2011.

3/4.6.5 CONTAINMENT ENCLOSURE BUILDING

3/4.6.5.1 CONTAINMENT ENCLOSURE EMERGENCY AIR CLEANUP SYSTEM (CEEACS)

BACKGROUND

The CEEACS is designed to maintain a negative pressure of greater than or equal to 0.25 inches of water, following a design basis accident, in the annular region defined by the containment structure and the containment enclosure, as well as in the additional building volumes associated with the electrical penetration areas, mechanical piping penetration area and engineered safeguard equipment cubicles. Any fission products leaking from these systems and from the primary containment will be retained in these areas and eventually processed through the filters.

The filter system consists of redundant filter trains, fans, dampers and controls and a common ductwork system. The air flow required to maintain a negative pressure in the containment enclosure building is passed through demisters, which also function as prefilters, and through HEPA filters located both upstream and downstream of the carbon filter prior to exhausting through the plant vent. A ductwork cross-connection is provided between the two filter trains at a point between the downstream HEPA filter and the fan inlet. Should the operating fan fail, this cross-connection will insure a continued air flow by manual startup of the redundant fan. Each redundant filter train is complete, separate and independent from both electrical and control standpoints. Each filter train fan is supplied power from an independent power source.

APPLICABLE SAFETY ANALYSIS

During LOCA conditions, the CEEACS ensures that containment vessel leakage into the annulus and radioactive materials leaking from engineered safety features equipment, the electrical penetration areas, and the mechanical penetration tunnel, will be filtered through the HEPA filters and charcoal adsorber trains prior to discharge to the atmosphere. In the event of an accident requiring CEEACS operation, both of the redundant filter train fans will be automatically started on a "T" signal. One train of the CEEACS is required to draw down the entire containment enclosure area to a negative differential pressure of 0.25 inches of water. This differential pressure is required to be established between all areas that comprise the containment enclosure area and their external surroundings.

3/4.6.5.1 CONTAINMENT ENCLOSURE EMERGENCY AIR CLEANUP SYSTEM (CEEACS) (continued)

Analysis has shown that one containment enclosure exhaust filter fan is capable of drawing down the entire containment enclosure area to the design negative differential pressure in less than eight minutes after the initiation of a design basis LOCA. This analysis takes into account the engineered safety feature actuation system signal delay time, delay time for the diesel generator to supply power in the event of a simultaneous loss of offsite power, and the time for the filter fan to come up to speed.

The CEEACS satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

One train of the CEEACS is required to maintain a negative pressure within the containment enclosure following an accident, to remove and retain airborne particulates and radioactive iodine, and to exhaust filtered air to the unit plant vent. Two trains of the CEEACS must be OPERABLE to ensure that at least one train will operate assuming that the other train is disabled by a single active failure. When the LCO for the CEEACS is not met, it is not necessary to declare LCO 3.6.5.2 for containment enclosure building integrity not met.

The CEEACS also provides cooling to the following areas and equipment during normal and emergency operation: containment enclosure ventilation equipment area, the charging pumps, safety injection pumps, residual heat removal pumps, containment spray pumps, and the mechanical penetration area. However, the cooling function is not associated with this TS, but rather is controlled under Technical Requirement 24, Area Temperature Monitoring.

The components associated with this TS include those dampers, fans, filters, etc., and required ductwork and instrumentation that evacuate or isolate areas, route air, and filter the exhaust prior to discharge to the environment. Included among these components are:

- Containment enclosure cooling fans (EAH-FN-5A and 5B)
- Containment enclosure ventilation area return fans (EAH-FN-31A and 31B)
- Containment enclosure emergency exhaust fans (EAH-FN-4A and 4B)
- Charging pump room return air fans (EAH-FN-180A and 180B)
- Containment enclosure emergency clean up filters (EAH-F-9 and F-69)
- PAB / CEVA isolation dampers (PAH-DP-35A, 36A, 35B, and 36B)

3/4.6.5.1 CONTAINMENT ENCLOSURE EMERGENCY AIR CLEANUP SYSTEM (CEEACS) (continued)

APPLICABILITY

In MODES 1, 2, 3, and 4, a design basis accident (DBA) could lead to fission product release to containment that leaks to the containment enclosure building. In MODES 5 and 6, the probability and consequences of a DBA are low due to the pressure and temperature limitations in these MODES. Under these conditions, the CEEACS is not required to be OPERABLE.

ACTION

The Action requires that with one CEEACS train inoperable, the inoperable train must be restored to OPERABLE status within 7 days. The 7 day completion time considers the availability of the OPERABLE redundant CEEACS train and the low probability of a design basis accident occurring during this period. If the CEEACS train cannot be restored to OPERABLE status within the 7 days, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 5 within the following 30 hours. The completion times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

3/4.6.5.2 CONTAINMENT ENCLOSURE BUILDING INTEGRITY

BACKGROUND

The containment enclosure building is a reinforced concrete right cylindrical structure with a hemispherical dome that is located outside and surrounds the containment building. This structure provides leak protection for the containment and protects it from certain loads (normal loads, loads due to severe and extreme environmental conditions, and abnormal loads). The space between the containment and the enclosure building is maintained at a slight negative pressure during accident conditions. All joints and penetrations are sealed to ensure air tightness.

Without containment enclosure building integrity, the containment building spray (CBS) system provides additional defense-in-depth for accidents that credit the containment enclosure building. The CBS system functions to remove iodine and reduce containment pressure, which reduces containment leakage to the containment enclosure. As a result, the CBS system reduces dose consequences from a release from the primary containment. When Action a or b is entered, the plant can continue to operate at power if at least one CBS train is operable in accordance with TS 3.6.2.1.

3/4.6.5.2 CONTAINMENT ENCLOSURE BUILDING INTEGRITY (continued)

APPLICABLE SAFETY ANALYSES

The function of the containment enclosure building is to collect any fission products that leak from the primary containment structure into the containment enclosure and contiguous areas following a LOCA. The containment enclosure provides a barrier between the containment and the environment to control all leakage out from the containment boundary. Containment enclosure building integrity ensures that the release of radioactive materials from the primary containment atmosphere will be restricted to those leakage paths and associated leak rates assumed in the safety analyses. This restriction, in conjunction with operation of the containment enclosure emergency air cleanup system (CEEACS), will limit radiation dose to within the dose guideline values of 10 CFR 50.67 during accident conditions.

The containment enclosure building satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

Containment enclosure building integrity must be maintained to limit the release of radioactive materials from the primary containment atmosphere to those leakage paths and associated leak rates assumed in the safety analyses. Containment enclosure building integrity exists when (1) each door in each access opening is closed except when the access opening is being used for normal transit entry and exit, (2) the sealing mechanism associated with each containment enclosure building penetration (e.g., welds, bellows, or O-rings) is OPERABLE, and (3) the containment enclosure building functions as designed to maintain the required negative pressure.

APPLICABILITY

Maintaining containment enclosure building integrity prevents leakage of radioactive material from the enclosure building. Radioactive material may enter the containment enclosure building from the containment following a DBA. Therefore, containment enclosure integrity is required in MODES 1, 2, 3, and 4 when a DBA could release radioactive material to the containment atmosphere. In MODES 5 and 6, the probability and consequences of these events are low due to the Reactor Coolant System temperature and pressure limitations in these MODES. Therefore, containment enclosure building integrity is not required in MODE 5 or 6.

3/4.6.5.2 CONTAINMENT ENCLOSURE BUILDING INTEGRITY (continued)

ACTIONS

A Note states that entry into the Actions is not required when an access opening (containment enclosure boundary door) is being used for normal transit entry and exit. This provision provides an exception to TS 3.0.1 when containment enclosure integrity is not maintained while an access door is open for normal transit. This note is consistent with SR 4.6.5.2.a, which requires each containment enclosure boundary door to be closed except during normal transit entry and exit.

Action a.

Action a addresses a loss of containment enclosure building integrity for reasons other than provided in Action b. For example, this action is applicable when performing preventative or corrective maintenance on the containment enclosure building boundary, including containment enclosure building penetration seals, dampers and access doors, that results in a failure to maintain containment enclosure building integrity. The containment enclosure building access openings contain a single door, so opening a door causes a loss of containment enclosure integrity.

Containment building enclosure integrity must be restored within 12 hours. Twelve hours is a reasonable completion time considering the limited leakage design of containment, the low probability of a DBA occurring during this time, and the time required to repair a containment enclosure building door.

Action b.

Action b addresses the condition in which an OPERABLE containment enclosure boundary door is held open to support movement of equipment through the access opening, or routing hoses, cables, etc., through the access opening. Thus, this action applies when containment enclosure building integrity is not maintained due to an open access door for equipment ingress and egress because the doorway must be maintained open, i.e., obstructed, for equipment, cables, hoses, etc., such that it cannot be immediately closed. Additionally, pressure boundary seals must also be intact to maintain the integrity of the containment enclosure. Action b does not apply to normal transit entry and exit.

Action b requires the availability of a dedicated individual with a preplanned method to rapidly close the containment enclosure boundary door in the event of actuation of the CEEACS. The dedicated individual must be stationed at the door and have continuous communications capability with the control room. Hoses and cables running through the access opening must employ a means that allows prompt removal of the obstruction to permit closure of the door without delay.

3/4.6.5.2 CONTAINMENT ENCLOSURE BUILDING INTEGRITY (continued)
Containment building enclosure integrity must be restored within 24 hours. Twenty-four hours is a reasonable completion time considering the limited leakage design of containment, the low probability of a DBA occurring during this time, and the availability of a dedicated individual to close the containment enclosure boundary door.

SURVEILLANCE REQUIREMENTS

SR 4.6.5.2.a

The containment enclosure boundary doors are normally maintained closed except when the access opening is being used for entry and exit. Verifying containment enclosure building integrity involves confirming that the doors are closed except during normal transit entry and exit. Normal transit includes opening doors as necessary to permit the movement of people and equipment through the doorway. This may also include opening doors to test actuation of door alarms. Propping open a door and obstructing the doorway with equipment, cables, hoses, etc., such that it cannot be immediately closed is not normal transit entry and exit. Additionally, pressure boundary seals must also be intact to maintain the integrity of the containment enclosure.

SR 4.6.5.2.b

The CEEACS is used to establish a negative pressure in the containment enclosure building. SR 4.6.5.2 verifies containment enclosure building integrity by drawing down the containment enclosure building to a negative pressure greater than or equal to 0.25 inch Water Gauge using one train of CEEACS within four minutes after a start signal to ensure that the building can meet its design negative pressure in less than eight minutes following the initiation of a LOCA. Inoperability of the containment enclosure building does not by itself render the CEEACS inoperable. Therefore, the Action of TS 3.6.5.1 (CEEACS) is not required to be entered solely due to a failure to maintain containment enclosure building integrity.

Since this SR is a containment enclosure building boundary integrity test, it does not need to be performed at each surveillance interval with each CEEACS train. The CEEACS train used for this SR is scheduled on a STAGGERED test basis to ensure that either train will perform the test. The primary purpose of this SR is to ensure containment enclosure building integrity. The secondary purpose of this SR is to ensure that the CEEACS train used for the test functions as designed. Inoperability of the CEEACS train does not necessarily constitute a failure of this SR relative to containment enclosure building integrity.

3/4.6.5.3 CONTAINMENT ENCLOSURE BUILDING STRUCTURAL INTEGRITY

This limitation ensures that the structural integrity of the containment enclosure building will be maintained comparable to the original design standards for the life of the facility.

The function of the containment enclosure building is to collect any fission products that leak from the primary containment structure into the containment enclosure and contiguous areas following an accident. The containment enclosure provides a low leakage rate barrier between the containment and the environment. Structural integrity of the containment enclosure building is necessary to prevent leakage of radioactive materials from the containment enclosure building. A visual inspection of the exposed interior and exterior concrete surfaces of the containment enclosure structure in accordance with the Containment Leakage Rate Testing Program, is sufficient to demonstrate this capability.