NEXTera® ENERGY SEABROOK

October 7, 2016
10 CFR 54

SBK-L-16146
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
Sixth Annual Update to the
NextEra Energy Seabrook License Renewal Application

References:

1. NextEra Energy Seabrook, LLC letter SBK-L-10077, "Seabrook Station Application for Renewed Operating License," May 25, 2010. (Accession Number ML101590099)

2. NextEra Energy Seabrook, LLC letter SBK-L-11773, "First Annual Update to the Seabrook License Renewal Application," August 25, 2011. (Accession Number ML11241A142)

3. NextEra Energy Seabrook, LLC letter SBK-L-12186, "Second Annual Update to the Seabrook License Renewal Application," September 18, 2012. (Accession Number ML12268A171)

4. NextEra Energy Seabrook, LLC letter SBK-L-13115, "Third Annual Update to the Seabrook License Renewal Application," July 2, 2013. (Accession Number ML13189A197)

5. NextEra Energy Seabrook, LLC letter SBK-L-13183, "Clarification to the Third Annual Update Provided in SBK-L-13115," October 21, 2013. (Accession Number ML13189A197)

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6. NextEra Energy Seabrook, LLC letter SBK-L-14173, "Fourth Annual Update to the Seabrook License Renewal Application," October 2, 2014. (Accession Number ML14282A023)

7. NextEra Energy Seabrook, LLC letter SBK-L-15170, "Fifth Annual Update to the Seabrook License Renewal Application," September 18, 2015. (Accession Number ML15271A161)

In Reference 1, NextEra Energy Seabrook, LLC 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.

The License Renewal Rule, 10 CFR 54.21(b) requires that each year following submittal of a license renewal application (LRA), and at least 3 months before scheduled completion of the NRC review, an update to the LRA must be submitted that identifies any change to the current licensing basis (CLB) of the facility that materially affects the content of the LRA including the Updated Final Safety Analysis Report Supplement.

In accordance with the License Renewal Rule, NextEra Energy Seabrook, LLC has performed a sixth annual review of CLB changes since the submittal of the LRA, to determine whether any sections of the LRA were affected by these changes. The first, second, third, fourth and fifth annual review results are documented in References 2, 3, 4, 5, 6 and 7. The sixth annual review did identify changes to the current licensing basis (CLB) of the facility that materially affected the content of the LRA. Provided in this Supplement are changes to the LRA. To facilitate understanding, the changes are explained, and where appropriate, portions of the LRA are repeated with the change highlighted by strikethroughs for deleted text and bolded italics for inserted text.

This letter contains no revised or new commitments.

If there are any questions or additional information is needed, please contact Mr. Edward J. Carley, Engineering Supervisor - License Renewal, at (603) 773-7957.

If you have any questions regarding this correspondence, please contact Mr. Ken Browne, Licensing Manager, at (603) 773-7932.

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

Executed on October 7, 2016

Sincerely,

Eric McCartney
Site Vice President
NextEra Energy Seabrook, LLC

Enclosures:

Enclosure 1- Sixth Annual Update to the NextEra Energy Seabrook Station License Renewal Application

cc:	D. H. Dorman	NRC Region I Administrator
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Enclosure 1 to SBK-L-16146

Sixth Annual Update to the NextEra Energy Seabrook Station License Renewal Application

During the fall 2015 refueling outage (OR17), NextEra implemented the proactive mitigation of primary water stress corrosion cracking (PWSCC) induced cracking through the Mechanical Stress Improvement Process on the remaining seven reactor vessel nozzle dissimilar metal welds. LRA sections 4.3.6, 4.7.3 and 4.7.13 related to Time Limited Aging Analysis (TLAA) have been revised as follows to include the performance of this activity.

4.3.6 ABSENCE OF TLAAS FOR FATIGUE CRACK GROWTH, FRACTURE MECHANICS STABILITY, OR CORROSION ANALYSES SUPPORTING REPAIR OF ALLOY 600 MATERIALS

Summary Description

Both Alloy 600 base material and Alloy 82/182 weld material have exhibited susceptibility to primary water stress corrosion cracking (PWSCC). Evaluations of these effects, or analyses in support of repairs to affected locations, can be TLAAs.

Analysis

Pressurizer

The pressurizer contains Alloy 600 material only as Alloy 82/182 welds attaching the surge, spray, and relief valve nozzles to the safe ends, and the safe ends to the connecting piping. Complete Alloy 690 structural weld overlays were completed on all of these locations during Refueling Outage 12 (Spring 2008). The overlays were supported by fatigue crack growth analyses. These fatigue crack growth analyses were projected for a 60-year life, to the end of the period of extended operation, and are therefore not TLAAs.

No base-metal corrosion analyses exist for the pressurizer, since no half-nozzle or similar repairs have exposed the base metal to reactor coolant.

Reactor Vessel

A reactor vessel hot leg nozzle Alloy 600 weld was mitigated through Mechanical Stress Improvement Process (MSIP) repair during Outage 13 (Fall 2009). 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. ***During Outage 17 (Fall 2015), the proactive mitigation of PWSCC induced cracking on the remaining seven reactor vessel nozzle dissimilar metal welds via MSIP was performed. This MSIP application was supported by a fatigue crack growth evaluation. This fatigue crack growth evaluation was projected to the end of the period of extended operation, and is therefore not a TLAA.***

There have been no other MSIP, Mechanical Nozzle Seal Assembly (MNSA), half-nozzle, or weld overlay repairs to reactor vessel Alloy 600 nozzle locations. Since there have been no MSIP, MNSA, half-nozzle, or weld overlay repairs to reactor vessel Alloy 600 nozzle locations, no other TLAA exists supporting their installation.

Steam Generators

The steam generator channel head drains (a/k/a bowl drains) contain Alloy 600 material. The channel head drains will be inspected periodically until the susceptible material is mitigated by replacing the alloy 600 welds.

Conclusion

No TLAAs for Fatigue Crack Growth, Fracture Mechanics Stability, or Corrosion Analyses Supporting Repair of Alloy 600 Materials exist for Seabrook Station. Components containing Alloy 600 material are monitored in accordance with B.2.1.1 ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD Program.

4.7.3 LEAK-BEFORE-BREAK ANALYSES

Summary Description

Title 10 Part 50 Appendix A, General Design "*Criteria for Nuclear Power Plants*" Criterion 4 of the Code of Federal Regulations allows for the use of leak-before-break (LBB) methodology for excluding the dynamic effects of postulated ruptures in reactor coolant system piping. The fundamental premise of the LBB methodology is that the materials used in nuclear power plant piping are sufficiently tough, that even a large through-wall crack would remain stable and would not result in a double-ended pipe rupture. Application of the LBB methodology is limited to those high-energy fluid systems not considered to be overly susceptible to failure from such mechanisms as corrosion, water hammer, fatigue, thermal aging or indirectly from such causes as missile damage or the failure of nearby components. The analyses involved with LBB are considered TLAAs.

Analysis

A LBB analysis was initially performed for Seabrook Station primary loop piping in 1984. To demonstrate the elimination of RCS primary loop pipe breaks, the following objectives had to be achieved:

- Demonstrate that margin exists between the "critical" crack size and a postulated crack that yields a detectable leak rate.

- Demonstrate that there is sufficient margin between the leakage through a postulated crack and the leak detection capability.
- Demonstrate margin on the applied load.
- Demonstrate that fatigue crack growth is negligible.

The initial analysis was reviewed to demonstrate compliance with LBB technology for Seabrook Station. Review of the 40-year LBB analysis considered input from the Stretch Power Uprate (2005) and the Mechanical Stress Improvement Process (MSIP) application at one of the reactor vessel primary hot leg nozzle locations (2009). Review for the period of extended operation was based on the same set of design transients as the original analyses; therefore the conclusions of the original evaluation remain valid for the 60-year period. ***An evaluation was performed for the proactive mitigation of PWSCC induced cracking on the remaining seven reactor vessel nozzle welds via MSIP (2015) and demonstrated that all the LBB recommended margins for the reactor coolant line piping continue to be satisfied after MSIP application.***

Plant specific geometry, operating parameters, loading, and material properties were used in the fracture mechanics evaluation. The mechanical properties were determined at operating temperatures. Since the piping systems also include cast austenitic stainless steel (CASS) piping components, fracture toughness considering thermal aging was determined for each affected component's heat of material for the fully aged condition.

Based on loading, pipe geometry, and fully aged fracture toughness considerations, enveloping governing locations were determined at which LBB crack stability evaluations were made. Through-wall flaw sizes were found which would cause a leak at a rate of ten (10) times the leakage detection system capability of the plant. Large margins for such flaw sizes were demonstrated against flaw instability. Finally, fatigue crack growth was shown not to be an issue for the reactor coolant system primary loop piping. The thermal transients used in the fatigue crack growth analysis were Seabrook Station design transients and projected cycles, which are reported in Table 4.3.1-3. The corresponding 60-year projected cycles, also shown in Table 4.3.1-3 are lower than the 40-year design values. Therefore, the numbers of design cycles assumed in the analysis bound the numbers of design cycles projected for 60 years of operation.

Disposition:

Validation, 10 CFR 54.21(c)(1)(i) – The analyses remain valid for the period of extended operation. The LBB review demonstrates that the previous LBB conclusions still remain valid, and the dynamic effects of the pipe rupture resulting from postulated breaks in the reactor coolant primary loop piping need not be considered in the Seabrook Station design basis for the period of extended operation.

4.7.13 ABSENCE OF A TLAA FOR INSERVICE FLAW GROWTH ANALYSES THAT DEMONSTRATE STRUCTURAL STABILITY FOR 40 YEARS

Summary Description

Defects discovered by inservice inspection or component failures may be repaired or replaced to restore the basis of the original design analysis; may be repaired or replaced to a different configuration, or may be analyzed to confirm that the as-found condition is acceptable. For ASME components these activities are controlled by Section XI, *"Rules for Inservice Inspection of Nuclear Power Plant Components."* A flaw analysis of a Class 1 component usually requires a fatigue crack growth analysis, which is a TLAA if it qualifies the component for the plant design life.

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***s*** ~~was~~ ***were*** supported by fatigue crack growth ~~analysis~~ ***analyses. These*** fatigue crack growth ~~analysis~~ ***analyses*** ~~was~~ ***were*** 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.