NEXTera ENERGY SEABROOK

January 9, 2013

10 CFR 50.90

SBK-L-12279
Docket No. 50-443

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

Seabrook Station

Response to Request for Additional Information Regarding License Amendment Request to Change Applicability of Technical Specification Pressure – Temperature Limits

References:

1. NextEra Energy Seabrook, LLC letter SBK-L-11186, "Application to Revise the Applicability of the Reactor Coolant System Pressure – Temperature Limits and the Cold Overpressure Protection Setpoints," November 17, 2011

2. NRC Letter "Seabrook Station, Unit 1 – Request for Additional Information Regarding License Amendment Request to Change Applicability of Technical Specification Pressure – Temperature Limits to 23.7 Effective Full-Power Years (TAC ME7645)," July 25, 2012

In Reference 1, NextEra Energy Seabrook, LLC (NextEra) submitted a license amendment request that proposes to revise the applicability period for the technical specification pressure-temperature limits and cold overpressure protection system setpoints from 20 to 23.7 effective full-power years. In Reference 2, the NRC requested additional information in order to complete its review of the license amendment request. The Enclosure to this letter contains NextEra's response to the request for additional information.

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

A001
NRR

Sincerely,

NextEra Energy Seabrook, LLC

Thomas A. Vehec
Plant General Manager

Enclosure

cc: NRC Region I Administrator
J. G. Lamb, NRC Project Manager, Project Directorate I-2
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AFFIDAVIT

SEABROOK STATION UNIT 1

Facility Operating License NPF-86
Docket No. 50-443

Response to Request for Additional Information Regarding License Amendment Request to Change Applicability of Technical Specification Pressure – Temperature Limits

I, Thomas A. Vehec, Plant General Manager of NextEra Energy Seabrook, LLC hereby affirm that the information and statements contained within this response to the request for additional information 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
9 day of January, 2013

Notary Public

Thomas A. Vehec
Plant General Manager

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

ENCLOSURE

Response to Request for Additional Information Regarding License Amendment Request to Change Applicability of Technical Specification Pressure – Temperature Limits

NRC Request for Additional Information (RAI)

P-T limit calculations for ferritic RCPB components that are not RV beltline shell materials may define P-T curves that are more limiting than those calculated for the RV beltline shell materials. This may be due to the following factors:

1. *RV nozzles, penetrations, and other discontinuities have complex geometries that may exhibit significantly higher stresses than those for the RV beltline shell region. These higher stresses can potentially result in more restrictive P-T limits, even if the reference temperature (RT_{NDT}) for these components is not as high as that of RV beltline shell materials that have simpler geometries.*

2. *Ferritic RCPB components that are not part of the RV may have initial RT_{NDT} values, which may define a more restrictive lowest operating temperature in the P -T limits than those for the RV beltline shell materials.*

Describe how the P-T limit curves, and the methodology used to develop these curves, considered all RV materials (beltline and non-beltline) and the lowest service temperature of all ferritic RCPB materials, consistent with the requirements of 10 CFR Part 50, Appendix G.

NextEra's Response to RAI

Background

The Seabrook Unit 1 20 effective full power year (EFPY) pressure – temperature (P-T) limits in WCAP-15745 [Reference 1] were developed using the methodology described in Topical Report WCAP-14040-A, Revision 2, "Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves" [Reference 2]. The applicability of the current 20 EFPY P-T limits was revised as a result of an updated fluence evaluation that is contained in WCAP-17441-NP [Reference 3]. The Seabrook License Amendment Request (LAR) [Reference 5] documented that the limiting fluence of Reference 3 used for the 20 EFPY P-T limits would not be reached until 23.7 EFPY. WCAP-17441-NP was provided in Reference 9. No other changes have occurred in the plant that would affect the P-T limits.

For Westinghouse nuclear steam supply systems, Topical Report WCAP-14040-A describes the methodology that is used to comply with the requirements of 10 CFR 50 Appendix G, "Fracture Toughness Requirements" [Reference 4]. The pressurizer and steam generators were designed and fabricated to meet the requirements of the 1971 Edition of the ASME Code, Section III through Summer 1973 Addenda. The reactor vessel was designed and fabricated to meet the requirements of the 1972 Edition of the ASME Code, Section III. Since only the reactor vessel (RV) undergoes neutron embrittlement, the RV beltline region is the most limiting RCS

component. Therefore, the methodology in WCAP-14040 only addressed the RV beltline region of the RCS as the most liming for the P-T limits. The NRC Safety Evaluation (SE) for this topical report states, "We find the report acceptable for referencing in the administrative controls section of the technical specifications for the license amendment application to the extent specified and under the limitations delineated in the report and the associated NRC safety evaluation, which is enclosed. The safety evaluation defines the basis for acceptance of the report." The SE further states, "The staff finds the WCAP-14040 methodology consistent with Appendix G to Section III of the ASME Code and SRP Section 5.3.2." and "T is the metal temperature and RT_{NDT} is the ART value of the limiting vessel material" confirming that the reactor vessel is the limiting component evaluated in the development of the P-T limits. Table 1 of the NRC SE provides requirements regarding the fluence methodology, surveillance capsule program requirements, LTOPS requirements, ART calculation, and 10 CFR 50 Appendix G temperature requirements which have all been addressed in WCAP-15745 consistent with the NRC SE.

NRC RAI

P-T limit calculations for ferritic RCPB components that are not RV beltline shell materials may define P-T curves that are more limiting than those calculated for the RV beltline shell materials. This may be due to the following factors:

NRC Factor #1

RV nozzles, penetrations, and other discontinuities have complex geometries that may exhibit significantly higher stresses than those for the RV beltline shell region. These higher stresses can potentially result in more restrictive P-T limits, even if the reference temperature (RT_{NDT}) for these components is not as high as that of RV beltline shell materials that have simpler geometries.

Response to NRC Factor #1

WCAP-14040, Revision 2 did not consider the embrittlement of ferritic materials in the area adjacent to the beltline, specifically the stressed nozzles. The nozzles are the most highly stressed ferritic component outside the beltline region of the reactor vessel. This issue is addressed below.

For the Seabrook Unit 1 P-T limit EFPY applicability LAR, the updated RV nozzle fluence in WCAP-17441-NP at 55 EFPY for the lowest extent of the inlet nozzles is 1.02E+17 n/cm^2 (E>1.0 MeV) and 5.70E+16 n/cm^2 for the outlet nozzles. These fluence values bound the nozzle fluence values for the 23.7 EFPY P-T limit curves. A calculation of the Seabrook nozzle adjusted reference temperature values and nozzle to upper shell welds using the bounding 55 EFPY fluence, Cu, Ni and initial RT_{NDT} values are shown in Table 1. The highest nozzle ¼ T Adjusted Reference Temperatures (ART) are 9.8°F for the inlet nozzles and 13.5°F for the outlet nozzles, as calculated per Regulatory Guide 1.99, Revision 2. The ART values for the nozzle to shell welds as shown in the table below are not limiting due to the low initial RT_{NDT} generic value for Linde 0091 as well as the lack of a stress concentration factor that is present at the forged nozzle corner. The stress intensity factor to be applied to the nozzle to shell welds is the

same as that of the beltline with cylindrical shell geometry. Therefore, an ART comparison proves that the beltline is limiting since the ART of the limiting component is 109°F for the lower shell plate R-1808-1 of which was used in the calculation of the P-T limits of Reference 1 (compared to 5.0 °F for the inlet nozzle to upper shell welds and -5.8 °F for the outlet nozzle to upper shell welds at 55 EFPY as shown in Table 1).

Table 1
Seabrook Unit 1 RV ART Values for the ¼ T Location at 55 EFPY Fluence

LOCATION	Cu Wt.%	Ni Wt.%	TABLE CF	INITIAL RT_{NDT}	σ_i	σ_Δ	MARGIN	55 EFPY SURFACE FLUENCE n/cm² (E>1MeV)	55 EFPY ¼T FLUENCE n/cm2 (E>1MeV)	55 EFPY ¼T ΔRT_{NDT}	55 EFPY ART @ ¼T
Inlet Nozzle bounding case	.10	.89	67°F	0°F	0°F	2.4°F	4.9°F	1.02E+17	5.62E+16	4.91°F	9.8°F
Outlet Nozzle bounding case	.40*	.67	240°F	-10 °F	0°F	5.9°F	11.8°F	5.70E+16	3.31E+16	11.8°F	13.5°F
Inlet Nozzle to Upper Shell welds (Linde 0091)	0.35*	1.00*	272°F	-56 °F	17°F	10.5°F	40.0°F	1.02E+17	6.1E+16	21.0°F	5.0°F
Outlet Nozzle to Upper Shell welds (Linde 0091)	0.35*	1.00*	272°F	-56 °F	17°F	6.8 °F	36.6°F	5.70E+16	3.4E+16	13.6°F	-5.8°F

* Note the Cu and Ni values are an extremely conservative estimates based on the highest Cu or generic Cu and Ni values reported in Regulatory Guide 1.99, Table 2. These values are not expected to be representative of the actual Cu and Ni values and lower values could be justified.

A calculation of the Seabrook nozzle cooldown P-T limits was completed, using the inlet and outlet nozzle Adjusted Reference Temperature properties at 55 EFPY fluence to account for nozzle embrittlement. The stress intensity factor correlations used for the nozzle corners are consistent with the proposed Appendix G Code revision [Reference 7] and ORNL study [Reference 8]. The methodology used included postulating an inside surface ¼ T nozzle corner flaw along with calculating through-wall nozzle corner stresses for a cooldown rate of 100°F/hour.

The through-wall stresses at the nozzle corner cut were fitted based on a third-order polynomial of the form:

$$\sigma = A_0 + A_1 x + A_2 x^2 + A_3 x^3$$

Where:

σ = through-wall stress distribution

x = through-wall distance from inside surface

A_0, A_1, A_2, A_3 = coefficients of polynomial fit for the third-order polynomial, used in the stress intensity factor expression discussed below.

The stress intensity factors generated for a rounded nozzle corner for the pressure and thermal gradient were calculated based on the methodology provided in ORNL/TM-2010/246 [Reference 8]. The SIF expression for a rounded corner is:

$$K_I = \sqrt{\pi a}\left[0.706A_o + 0.537\left(\frac{2a}{\pi}\right)A_1 + 0.448\left(\frac{a^2}{2}\right)A_2 + 0.393\left(\frac{4a^3}{3\pi}\right)A_3\right]$$

Where:

K_I = stress intensity factor based on a rounded corner

a = crack depth at the nozzle corner, use with ¼ T (25% of the wall thickness)

It should be noted that an outside surface nozzle flaw was not considered. The pressure stress is significantly lower at the outside surface than the inside surface. Further, a stress intensity factor correlation does not exist to consider an outside surface nozzle corner flaw. Therefore, a heatup nozzle P-T limit curve is not provided, as it would be less limiting than the nozzle P-T limit curve in Figures 1 and 2. The resulting nozzle P-T curves, as provided in Figures 1 and 2 were less limiting than that of the 20 EFPY (revised to 23.7 EFPY) cooldown limits in the subject P-T limits contained in WCAP-15745. This demonstrates that the nozzle P-T limits are less limiting and the RV beltline is controlling.

As discussed previously, the nozzles are the most highly stressed ferritic component outside the beltline region of the reactor vessel. Consequently, the inlet / outlet nozzles are the most limiting reactor vessel components with respect to the effects of stress concentration and resulting stress levels and are controlling with respect to any other vessel penetrations.

NRC Factor #2

Ferritic RCPB components that are not part of the RV may have initial RT_{NDT} values, which may define a more restrictive lowest operating temperature in the P-T limits than those for the RV beltline shell materials.

Response to NRC Factor #2

The Lowest Service Temperature (LST), applicable to material for ferritic piping, pumps and valves with a nominal wall thickness of 2 ½ in. and less is RT_{NDT} +100°F [Reference 6]. Materials with nominal wall thicknesses greater than 2 ½ in. shall meet the requirements of NB-2331 unless a lower temperature is justified by the methods of Appendix G. The Seabrook reactor coolant system does not have ferritic materials in the piping, pumps or valves. Therefore,

the Lowest Service Temperature requirements of NB-2331 are not applicable to the Seabrook P-T limits.

The Seabrook steam generators and pressurizer were designed to the 1971 Edition of the ASME Code, Section III through Summer 1973 Addenda requirements. These components are original and have not been replaced. The original pressurizer and steam generators have not undergone neutron embrittlement that would affect P-T limits. Therefore further consideration of these components for pressure-temperature limits is not required.

RAI

Describe how the P-T limit curves and the methodology used to develop these curves, considered all RV materials (beltline and non-beltline) and the lowest service temperature of all ferritic RCPB materials, consistent with the requirements of 10CFR Part 50, Appendix G.

Response to RAI

Nozzle embrittlement has been addressed for the 20 EFPY (revised to 23.7 EFPY) Seabrook Unit 1 P-T limits and shown to be less limiting than the 20 EFPY P-T limits of Reference 1. Furthermore, the lowest service temperature requirements of Reference 6 are not applicable to Seabrook. The other ferritic components of the RCS, namely the steam generator and pressurizer, do not require further evaluation since they were designed to the requirements of ASME Section III and have not received neutron embrittlement. Therefore, the P-T limit applicability evaluation in Seabrook Unit 1 LAR is consistent with the requirements of 10 CFR 50, Appendix G.

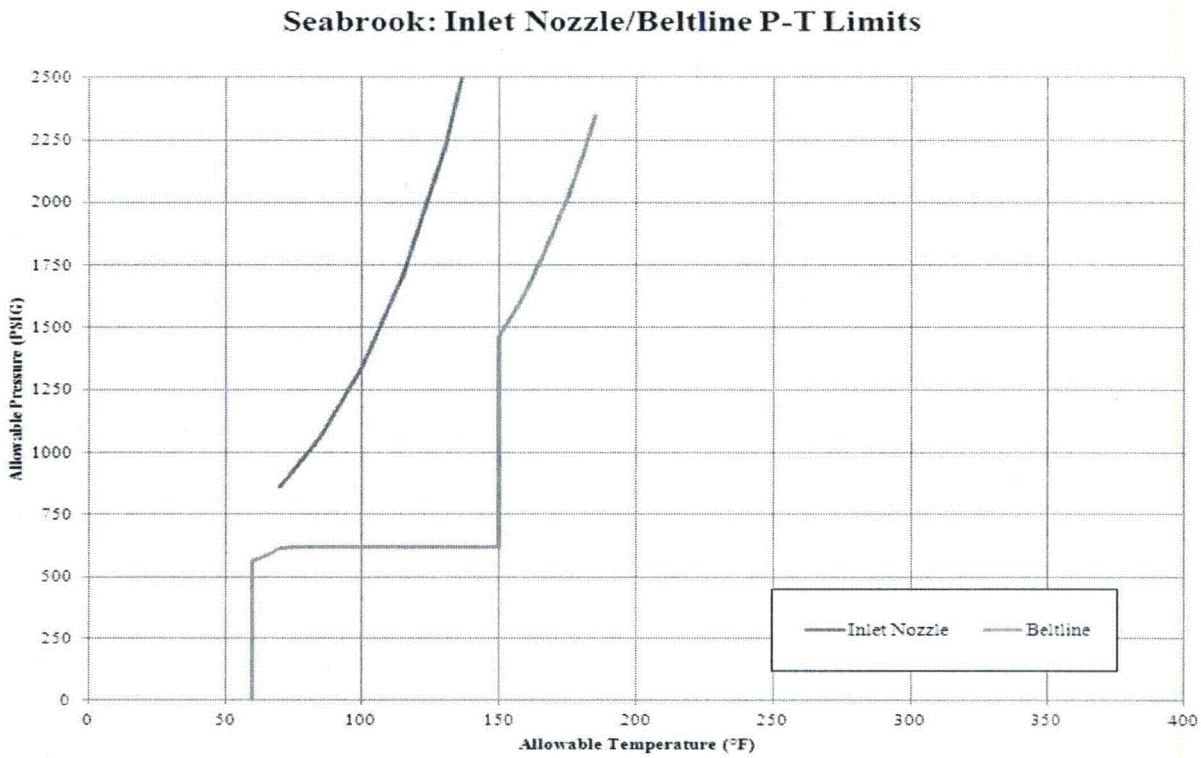

Figure 1: Comparison of Seabrook WCAP-15745 P-T Limits to Inlet Nozzle Limits

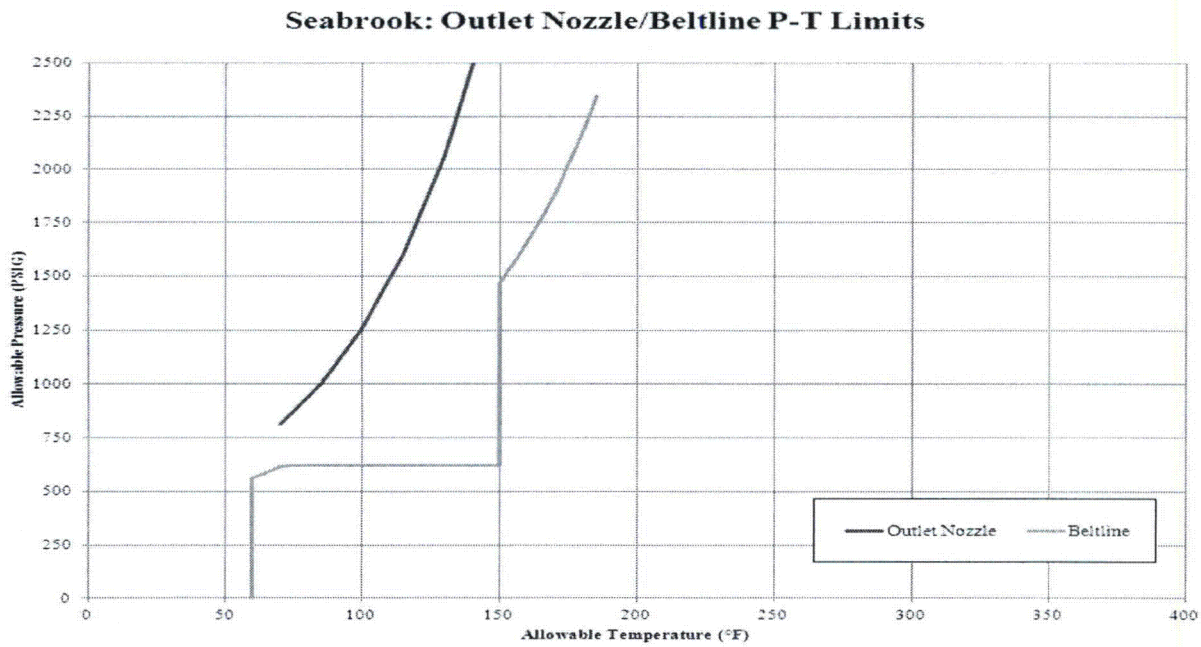

Figure 2: Comparison of Seabrook WCAP-15745 P-T Limits to Outlet Nozzle Limits

References

1. WCAP-15745, Revision 0, "Seabrook Unit 1 Heatup and Cooldown Limit Curves for Normal Operation," T. J. Laubham, December 2001.
2. WCAP-14040-A, Revision 2, "Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves," J. D. Andrachek, et al., January 1996.
3. WCAP-17441-NP, Revision 0, "Seabrook Unit 1 Heatup and Cooldown Limit Curves for Normal Operation," B. A. Rosier et al., October 2011.
4. Code of Federal Regulations, 10 CFR Part 50 Appendix G, "Fracture Toughness Requirements," U. S. Nuclear Regulatory Commission, Federal Register, Volume 60, No. 243, December 19, 1995.
5. SBK-L-11186, License Amendment Request 11-06, "License Amendment Request 11-06 - Application to Revise the Applicability of the Reactor Coolant System Pressure-Temperature Limits and the Cold Overpressure Protection Setpoints," P. Freeman, November 17, 2011, Accession No. ML11329A017.
6. 2004 Edition of the ASME B&PV Code Section III, Division I, NB-2332, "Material for Piping, Pumps, and Valves, Excluding Bolting Material."
7. ASME PVP2011-57015, "Additional Improvements to Appendix G of ASME Section XI Code for Nozzles," G. Stevens, H. Mehta, T. Griesbach, D. Sommerville, July 2011.
8. ORNL/TM-2010/246, "Stress and Fracture Mechanics Analyses of Boiling Water Reactor and Pressurized Water Reactor Pressure Vessel Nozzles – Revision 1," June 2012.
9. SBK-L-12256, Supplemental Information for "License Amendment Request 11-06 Application to Revise the Applicability of the Reactor Coolant System Pressure - Temperature Limits and the Cold Overpressure Protection Setpoints," M. O'Keefe, December 3, 2012, Accession No. ML12341A095.