NEXTera ENERGY SEABROOK

L-2023-012
10 CFR 50.90
March 15, 2023

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

RE: Seabrook Station
Docket No. 50-443
Renewed Facility Operating License No. NPF-86

License Amendment Request 23-01, Remove Period of Applicability (POA) from Pressure-Temperature Limits (PTL) and Low Temperature Overpressure Protection (LTOP) Curves

Reference:

1. NextEra Energy Seabrook, LLC, letter SBK-L-21106, Transmittal of WCAP-18607-NP, Analysis of Capsule X from the NextEra Energy Seabrook Unit 1 Reactor Vessel Radiation Surveillance Program, September 30, 2021 (ADAMS Accession No. ML21277A388)

Pursuant to 10 CFR 50.90, NextEra Energy Seabrook, LLC (NextEra) hereby requests an amendment to Renewed Facility Operating License NPF-86 for Seabrook Station Unit 1 (Seabrook). The proposed license amendment modifies the Seabrook Technical Specifications (TS) by removing the specified period of applicability (POA) of 55 effective full-power years (EFPY) from the pressure-temperature limits (PTL) curves of Seabrook TS Figure 3.4-2, Reactor Coolant System Heatup Limitations - Applicable to 55 EFPY, and Figure 3.4-3, Reactor Coolant System Cooldown Limitations - Applicable to 55 EFPY, and from Figure 3.4-4, Maximum Allowable PORV Setpoints for Cold Overpressure Protection System. As reported in Reference 1, the requested amendment follows the latest reactor vessel peak fluence projections and updated POA based on surveillance capsule dosimetry obtained at 26.46 EFPY.

The current period of applicability for Seabrook TS Figure 3.4-2, Figure 3.4-3, and Figure 3.4-4 is being treated as a non-conservative TS and managed in accordance with Regulatory Guide (RG) 1.239, Licensee Actions to Address Nonconservative Technical Specifications (ADAMS Accession No. ML20294A510). Accordingly, this license amendment request is required to resolve a non-conservative TS and is not a voluntary request to change the Seabrook licensing basis.

The enclosure to this letter provides a description and assessment of the proposed changes. Attachment 1 to the enclosure provides the existing TS pages marked to show the proposed changes. Attachment 2 provides the existing TS Bases pages marked up to show the proposed changes. The TS Bases page changes are provided for information only and will be implemented in accordance with the Seabrook TS Bases Control Program upon implementation of the proposed license amendment.

NextEra has determined that the proposed change does not involve a significant hazards consideration pursuant to 10 CFR 50.92(c), and there are no significant environmental impacts associated with the change. The Seabrook Onsite Review Group (ORG) has reviewed the proposed license amendment. In accordance with 10 CFR 50.91, a copy of this application, with attachments, is being provided to the designated New Hampshire official.

NextEra requests that the proposed change is processed as a normal license amendment request, with approval within one year of receipt. Once approved, the amendment shall be implemented within 90 days.

This letter contains no new or revised regulatory commitments.

Should you have any questions regarding this submission, please contact Mr. Kenneth Mack, Fleet Licensing Manager, at 561-904-3635.

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

Executed on the 15th day of March 2023.

Sincerely,

for

Dianne Strand
General Manager, Regulatory Affairs

Enclosure
Attachments

cc: USNRC Region I Administrator
USNRC Project Manager
USNRC Senior Resident Inspector

Director Homeland Security and Emergency Management
New Hampshire Department of Safety
Division of Homeland Security and Emergency Management
Bureau of Emergency Management
33 Hazen Drive
Concord, NH 03305

Katharine Cederberg, Lead Nuclear Planner
The Commonwealth of Massachusetts
Emergency Management Agency
400 Worcester Road
Framingham, MA 01702-5399

Description and Assessment

Seabrook Station

License Amendment Request 23-01, Revise Pressure-Temperature Limits (PTL) Curve Service Period Based on Surveillance Capsule Dosimetry Analyses

ATTACHMENTS

1. Proposed Technical Specification Changes (mark-up)
2. Proposed Technical Specification Bases Changes (mark-up)

1.0 SUMMARY DESCRIPTION

NextEra Energy Seabrook, LLC (NextEra) requests an amendment to Renewed Facility Operating License NPF-86 for Seabrook Station Unit 1 (Seabrook). The proposed license amendment modifies the Seabrook Technical Specifications (TS) by removing the specified period of applicability (POA) of 55 effective full-power years (EFPY) from the pressure-temperature limits (PTL) curves of Seabrook TS Figure 3.4-2, Reactor Coolant System Heatup Limitations - Applicable to 55 EFPY, and Figure 3.4-3, Reactor Coolant System Cooldown Limitations - Applicable to 55 EFPY, and from Figure 3.4-4, Maximum Allowable PORV Setpoints for Cold Overpressure Protection System. As reported in Reference 6.1, the requested amendment follows the latest reactor vessel peak fluence projections and updated POA based on surveillance capsule dosimetry obtained at 26.46 EFPY.

2.0 DETAILED DESCRIPTION

2.1 System Design and Operation

10 CFR Part 50, Appendix A, General Design Criterion (GDC) 31, "Fracture Prevention of Reactor Coolant Pressure Boundary", requires that all components of the Reactor Coolant System (RCS) are designed to withstand the effects of cyclic loads due to system pressure and temperature changes, and include an adequate margin to brittle failure during normal operation and anticipated operational occurrences. Seabrook TS 3.4.9.1, Pressure/Temperature Limits, specify RCS heatup and cooldown pressure-temperature limits for normal operation which preclude operating conditions that might cause non-ductile failure of the reactor coolant pressure boundary (RCPB). Development of the pressure-temperature limits (PTL) curves for the Seabrook RCPB considers the vessel shell material with the highest reference temperature as well as other materials with structural discontinuities, and in particular, the reactor vessel nozzle materials. All ferritic components within the Seabrook RCPB, and the effects of neutron radiation, are considered in the development of the PTL curves for any materials that are projected to experience an end-of-license neutron exposure greater than $1x10^{17}$ neutrons per square centimeter (n/cm^2). The PTL curves meet the requirements of American Society of Mechanical Engineers (ASME) Code, Section III and Section XI, as required by 10 CFR Part 50, Appendix G, which requires the establishment of pressure-temperature limits based on specific material fracture toughness requirements. The Seabrook PTL curves account for margin in pressure and temperature instrument uncertainties.

The effect of neutron embrittlement on the material toughness is reflected by increasing the nil ductility reference temperature (RT_{NDT}) as exposure to neutron fluence increases. The actual shift in the RT_{NDT} is determined periodically by removing and evaluating irradiated reactor vessel material specimens as a part of the surveillance capsule dosimetry analysis required by 10 CFR 50, Appendix H, Reactor Vessel Material Surveillance Program Requirements. The most limiting RT_{NDT} at any period in the reactor's life, expressed as the period of applicability (POA) in units of effective full power years (EFPY), is the change in RT_{NDT} (ΔRT_{NDT}) due to the radiation exposure associated with that service period, along with a margin term, added to the initial RT_{NDT} (IRT_{NDT}) to arrive at an adjusted RT_{NDT} (ART). The operating PTL curves are adjusted, as necessary, based on the evaluation findings and in accordance with Regulatory Guide (RG) 1.99, Revision 2, Radiation Embrittlement of Reactor Vessel Materials. RG 1.99 describes an NRC approved method for calculating the effects of neutron radiation embrittlement at the reactor vessel 1/4T and 3/4T locations, where T is the vessel thickness at the beltline region measured from the clad/base metal interface. The most limiting ART values are used for generating the PTL curves in Seabrook TS Figures 3.4-2 and 3.4-3, as well as for projecting future peak fluence values used for scheduling future surveillance capsule evaluations.

10 CFR 50, Appendix G, also establishes limits for the low-temperature overpressure protection (LTOP) system, which serves to minimize the potential for challenging reactor vessel integrity when operating at or near reactor vessel ductility during cold shutdown, heatup, and cooldown operations. For Seabrook, the design basis transients applicable to these limiting conditions are (1) the start of a

single centrifugal charging pump into a water-solid RCS with letdown isolated, and (2) the start of an idle reactor coolant pump (RCP) with all RCS loops inactive and steam generator secondary side temperature 50°F hotter than the RCS primary side temperature. Seabrook TS 3.4.9.3, Overpressure Protection Systems, contains the LTOP system requirements and provides for the pressurizer power operated relief valves (PORVs) to mitigate low temperature overpressure transients to assure peak RCS pressure will not exceed 10 CFR 50, Appendix G, limits during operational transients. The PORV setpoints contained in Seabrook TS Figure 3.4-4, Maximum Allowable PORV Setpoints for Cold Overpressure Protection System, are based on the pressure-temperature limits established in accordance with 10 CFR 50, Appendix G.

2.2 Current Requirements / Description of the Proposed Changes

2.2.1 The Seabrook TS Index lists:

- Figure 3.4-2, Reactor Coolant System Heatup Limitations - Applicable to 55 EFPY, and
- Figure 3.4-3, Reactor Coolant System Cooldown Limitations - Applicable to 55 EFPY

The proposed change revises the Seabrook TS Index by removing "- Applicable to 55 EFPY" from the Figure 3.4-2 and Figure 3.4-3 listed titles. Attachment 1 to this amendment request provides the TS Index markup to show the proposed change.

2.2.2 Seabrook TS 3.4.9.1, Figure 3.4-2, Reactor Coolant System Heatup Limitations - Applicable to 55 EFPY, specifies:

- 55 EFPY as the POA for the ART limit material property basis,
- 55 EFPY as the POA for the RCS pressure versus temperature curves,
- 55 EFPY as the POA (aka service period) for the criticality limit based on the inservice hydrostatic test temperature (197°F)
- 55 EFPY as the POA identified in the TS Figure title.

The proposed change revises Seabrook TS 3.4.9.1, Figure 3.4-2, by removing 55 EFPY from the figure title and from all references in the figure drawing, and related editorial changes. Attachment 1 provides the Figure 3.4-2 markup to show the proposed change.

2.2.3 Seabrook TS 3.4.9.1, Figure 3.4-3, Reactor Coolant System Cooldown Limitations - Applicable to 55 EFPY, specifies:

- 55 EFPY as the POA for the limiting ART material property basis,
- 55 EFPY as the POA for the RCS pressure versus temperature curves,
- 55 EFPY as the POA identified in the TS Figure title.

The proposed change revises Seabrook TS 3.4.9.1, Figure 3.4-3, by removing 55 EFPY from the figure title and from all references in the figure drawing, and related editorial changes. Attachment 1 provides the Figure 3.4-3 markup to show the proposed change.

2.2.4 Seabrook TS 3.4.9.1, Figure 3.4-4, RCS Cold Overpressure Protection Setpoints, specifies:

- 55 EFPY as the POA for the PORV setpoint versus RCS temperature curve.

The proposed change revises Seabrook TS 3.4.9.1, Figure 3.4-4, by removing 55 EFPY from all references in the figure drawing, and related editorial changes. Attachment 1 provides the Figure 3.4-4 markup to show the proposed change.

2.3 Reason for the Proposed Change

The proposed change would eliminate the need for future license amendment requests to update the POA following 10 CFR 50, Appendix H, evaluations while continuing to limit plant operation within the PTL curves and LTOP system limits determined in accordance with 10 CFR 50, Appendix G.

3.0 TECHNICAL EVALUATION

In Reference 6.2, the NRC issued Seabrook Amendment No. 151, which revised TS 3.4.9.1, Reactor Coolant System Pressure/Temperature Limits, and TS 3.4.9.3, Overpressure Protection Systems, to include revised RCS heatup, cooldown, and pressure test operating requirements, and revised overpressure mitigation system requirements. The amendment authorized the PTL limit curves for 55 EFPYs of Seabrook operation. The amendment additionally revised the LTOP system requirements by establishing revised PORV setpoints based on the 55 EFPY PTL limits and by changing the RCS cold leg temperature at which the LTOP system must be operable. The amendment included the PTL curves of Seabrook TS Figure 3.4-2 and Figure 3.4-3, and the LTOP curve of Seabrook TS Figure 3.4-4 that are currently in use. In Reference 6.2, the NRC also granted an exemption from specific minimum temperature requirements of 10 CFR Part 50, Appendix G, Table 1, which allowed the use of an alternate methodology contained in WCAP-17444-NP, Revision 0, (Reference 6.3), in lieu of the Table 1 minimum temperature requirements. WCAP-17444-NP formed the bases for the PTL curves of Seabrook TS Figure 3.4-2 and Figure 3.4-3 and the LTOP system limits of Seabrook TS Figure 3.4-4 using ART values for the most limiting reactor pressure vessel beltline shell material. The application was based on the approved generic pressure-temperature limits methodology documented in WCAP-14040-A, Revision 4, (Reference 6.4) and the neutron transport evaluation methodologies of Regulatory Guide (RG) 1.190 (Reference 6.5).

In Reference 6.1, NextEra submitted the analysis of surveillance capsule 'X' from the Seabrook Unit 1 Reactor Vessel Radiation Surveillance Program, as required by 10 CFR 50, Appendix H. The results of the capsule X analysis concluded that the POA for the PTL curves in TS Figures 3.4-2 and 3.4-3 and the LTOP curve of TS Figure 3.4-4 will be reduced from 55 EFPY to 52.6 EFPY. Since the capsule X specimens were obtained at 26.52 EFPY, the PTL and LTOP curves currently in use remain conservative and valid for continued use for approximately another 25 EFPYs. However, the change in the POA resulting from the capsule X analysis renders non-conservative the POA currently specified in TS Figure 3.4-2, Figure 3.4-3, and Figure 3.4-4. The issue is being tracked in the Seabrook corrective action program (CAP) as a non-conservative TS whereby authorization to update the POA in TS Figure 3.4-2, Figure 3.4-3 and Figure 3.4-4 implements the final corrective action. In Reference 6.1, NextEra agreed to submit within one year, a license amendment request for an administrative change which replaces the current POA of 55 EFPY with 52.6 EFPY in the respective titles and notations of TS Figure 3.4-2, Figure 3.4-3, and Figure 3.4-4. In a February 23, 2022 pre-submittal meeting with the NRC (ADAMS Accession No. ML22040A212), NextEra agreed to submit the license amendment request in fourth quarter 2022. During subsequent discussion with the NRC, it was agreed that submittal in early 2023 would be acceptable.

In lieu of updating the POA from 55 EFPY to 52.6 EFPY based on the capsule X analysis, NextEra proposes to remove the POA from titles of Seabrook TS Figure 3.4-2 and Figure 3.4-3 and from the associated PTL and LTOP system limit curves specified in TS Figure 3.4-2, Figure 3.4-3, and Figure 3.4-4. NextEra believes the POA associated with TS Figure 3.4-2, Figure 3.4-3 and Figure 3.4-4 is appropriate for removal from the Seabrook TS because the POA does not satisfy the 10 CFR 50.36c(2)(ii) criteria for TS inclusion. Specifically, the POA is not instrumentation installed to detect and indicate in the control room significant abnormal degradation of the RCS pressure boundary, and thereby does not satisfy *Criterion 1*. The POA is not a process variable, design feature or operating restriction that is an initial condition assumed in any accident or transient analyses which challenges fission product barrier integrity, and thereby does not satisfy *Criterion 2*. The POA is not a structure, system or component (SSC) that is part of the primary success path to mitigate a design basis accident or transient challenging fission product barrier integrity, and thereby does not satisfy *Criterion 3*. Finally, the POA is not an SSC which operating experience or the Seabrook probabilistic risk assessment (PRA) has shown to be significant to public health and safety, and thereby does not satisfy *Criterion 4*. Based on the foregoing and consistent with the NRC Final Policy Statement on TS Improvements (Reference 6.6), the POA is appropriate for relocation to licensee control whereby future changes will be subject to regulatory change control requirements of 10 CFR 50.59. Moreover, there are no explicit regulations associated with the POA other than the 10 CFR 50, Appendix G, requirement authorizing reactor vessel operation within the service period (i.e., POA) for which the fracture toughness requirements therein are met. Given that 10 CFR 50, Appendix G, establishes overriding regulatory controls, NextEra believes retention of the POA in Seabrook TS Figure 3.4-2, Figure 3.4-3, and Figure 3.4-4 is unnecessary and proposes its removal to preclude the need for future amendment requests to revise the POA as a result of 10 CFR 50, Appendix H, related activities.

4.0 REGULATORY EVALUATION

4.1 Applicable Regulatory Requirements Criteria

- General Design Criteria (GDC) 31 states in part that the reactor coolant pressure boundary shall be designed with sufficient margin to assure that when stressed under operating, maintenance, testing, and postulated accident conditions, the boundary behaves in a nonbrittle manner and that the probability of rapidly propagating fracture is minimized.

- 10 CFR 50 Appendix G, Fracture Toughness Requirements, prescribes fracture toughness requirements for ferritic materials of pressure-retaining components of the reactor coolant pressure boundary, including applicable ASME Section XI, Appendix G limits.

- 10 CFR 50 Appendix H, Reactor Vessel Material Surveillance Program Requirements, prescribes material surveillance program requirements to monitor changes in the fracture toughness properties of ferritic materials in the reactor vessel resulting from neutron irradiation and the thermal environment.

- 10 CFR 50.36(c)(2)(ii) states that a limiting condition for operation must be established for each item meeting one or more of the four criteria specified therein.

- Regulatory Guide (RG) 1.99, Revision 2, describes general procedures for calculating the effects of neutron radiation embrittlement of the low-alloy steels currently used for light-water-cooled reactor vessels.

- RG 1.190, Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence, describes a methodology for determining the best-estimate neutron fluence experienced by materials in the reactor vessel beltline region and for determining the overall uncertainty associated with those values.

The proposed change complies with GDC 31, 10 CFR 50, Appendix G, 10 CFR 50, Appendix H, 10 CFR 50.36(c)(2)(ii), RG 1.99, Revision 2, and RG 1.190, consistent with regulatory requirements and guidelines. Therefore, all applicable requirements will continue to be satisfied upon implementation of the proposed license amendment.

4.2 No Significant Hazards Consideration Analysis

NextEra Energy Seabrook, LLC (NextEra) requests an amendment to Renewed Facility Operating License NPF-86 for Seabrook Station Unit 1 (Seabrook). The proposed license amendment modifies the Seabrook Technical Specifications (TS) by removing the specified period of applicability (POA) of 55 effective full-power years (EFPY) from the pressure-temperature limits (PTL) curves of

Seabrook TS Figure 3.4-2, Reactor Coolant System Heatup Limitations - Applicable to 55 EFPY, and Figure 3.4-3, Reactor Coolant System Cooldown Limitations - Applicable to 55 EFPY, and from Figure 3.4-4, Maximum Allowable PORV Setpoints for Cold Overpressure Protection System. As reported in Reference 6.1, the requested license amendment follows the latest reactor vessel peak fluence projections and updated POA based on surveillance capsule dosimetry obtained at 26.46 EFPY.

NextEra has evaluated if a significant hazards consideration is involved with the proposed amendment by focusing on the three standards set forth in 10 CFR 50.92, "Issuance of amendment," as discussed below:

(1) Does the proposed amendment involve a significant increase in the probability or consequences of an accident previously evaluated?

Response: No

The proposed change removes the POA from the Seabrook TS while retaining the 10 CFR 50, Appendix G, requirements for reactor pressure vessel operation within the applicable RCS pressure-temperature and over-pressure protection limits of the TS, which remain valid for another 25 EFPYs. The proposed change neither alters plant equipment nor the way in which plant equipment is operated or maintained, and thereby cannot increase the probability of any previously evaluated accident. The proposed change cannot affect the type or amount of effluent that can be released off-site or increase individual or cumulative occupational exposures, and thereby cannot increase the consequences of a previously evaluated accident.

Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

(2) Does the proposed amendment create the possibility of a new or different kind of accident from any accident previously evaluated?

Response: No

Removal of the POA from the Seabrook TS does not modify the RCS pressure-temperature and over-pressure protection limits of the TS, which remain valid for another 25 EFPYs. The proposed change neither installs new nor modifies existing plant equipment and thereby cannot introduce new equipment failure modes. The proposed change does not alter safety analysis assumptions, or create new accident initiators or precursors, and thereby cannot introduce a new or different type of accident.

Therefore, the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

(3) Does the proposed amendment involve a significant reduction in a margin of safety?

Response: No

Removal of the POA from the Seabrook TS does not affect the actual PTL and LTOP limit curves and thereby does not affect the operating margin associated with the RCS pressure-temperature and over-pressure protection limits. The proposed change does not modify any safety limits, limiting safety system settings, or safety analysis assumptions or inputs, and thereby cannot affect plant operating margins. The proposed change does not modify equipment credited in safety analyses, and thereby cannot affect the integrity of any radiological barrier.

Therefore, the proposed change does not involve a significant reduction in a margin of safety.

Based on the above, NextEra concludes that the proposed change presents no significant hazards consideration under the standards set forth in 10 CFR 50.92(c), and, accordingly, a finding of "no significant hazards consideration" is justified.

4.3 Conclusion

In conclusion, based on the considerations discussed above, (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

5.0 ENVIRONMENTAL CONSIDERATION

A review has determined that the proposed amendment would change a requirement with respect to installation or use of a facility component located within the restricted area, as defined in 10 CFR 20, or would change an inspection or surveillance requirement. However, the proposed amendment does not involve (i) a significant hazards consideration, (ii) a significant change in the types or a significant increase in the amounts of any effluents that may be released offsite, or (iii) a significant increase in individual or cumulative occupational radiation exposure. Accordingly, the proposed amendment meets the eligibility criterion for categorical exclusion set forth in 10 CFR 51.22(c)(9). Therefore, pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the proposed amendment.

6.0 REFERENCES

1. NextEra Energy Seabrook, LLC, letter SBK-L-21106, Transmittal of WCAP-18607-NP, Analysis of Capsule X from the NextEra Energy Seabrook Unit 1 Reactor Vessel Radiation Surveillance Program, September 30, 2021 (ADAMS Accession No. ML21277A388)

2. NRC letter to NextEra Energy Seabrook, LLC, Seabrook Station, Unit No. 1 - Issuance of Amendment Regarding License Amendment Request 14-04, Revised Reactor Coolant System Pressure/Temperature Limits Applicable for 55 Effective Full Power Years (TAC No. Mf4577), November 2, 2015 (ADAMS Accession No. 15096A255)

3. WCAP-17441-NP, Revision 0, Seabrook Unit 1 Heat-up and Cooldown Limit Curves for Normal Operation, October 2011, (ADAMS Accession No. ML12341A096)

4. WCAP-14040-A, Rev. 4, Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves, May 2004 (ADAMS Accession No. ML050120209).

5. Regulatory Guide (RG) 1.190, Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence, March 2001, (ADAMS Accession No. ML010890301)

6. NRC Final Policy Statement on Technical Specification Improvements for Nuclear Power Reactors, dated July 22, 1993 (58 FR 39132)

ATTACHMENT 1

PROPOSED TECHNICAL SPECIFICATION CHANGES (MARK-UP)

(5 pages follow)

3.0/4.0 LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

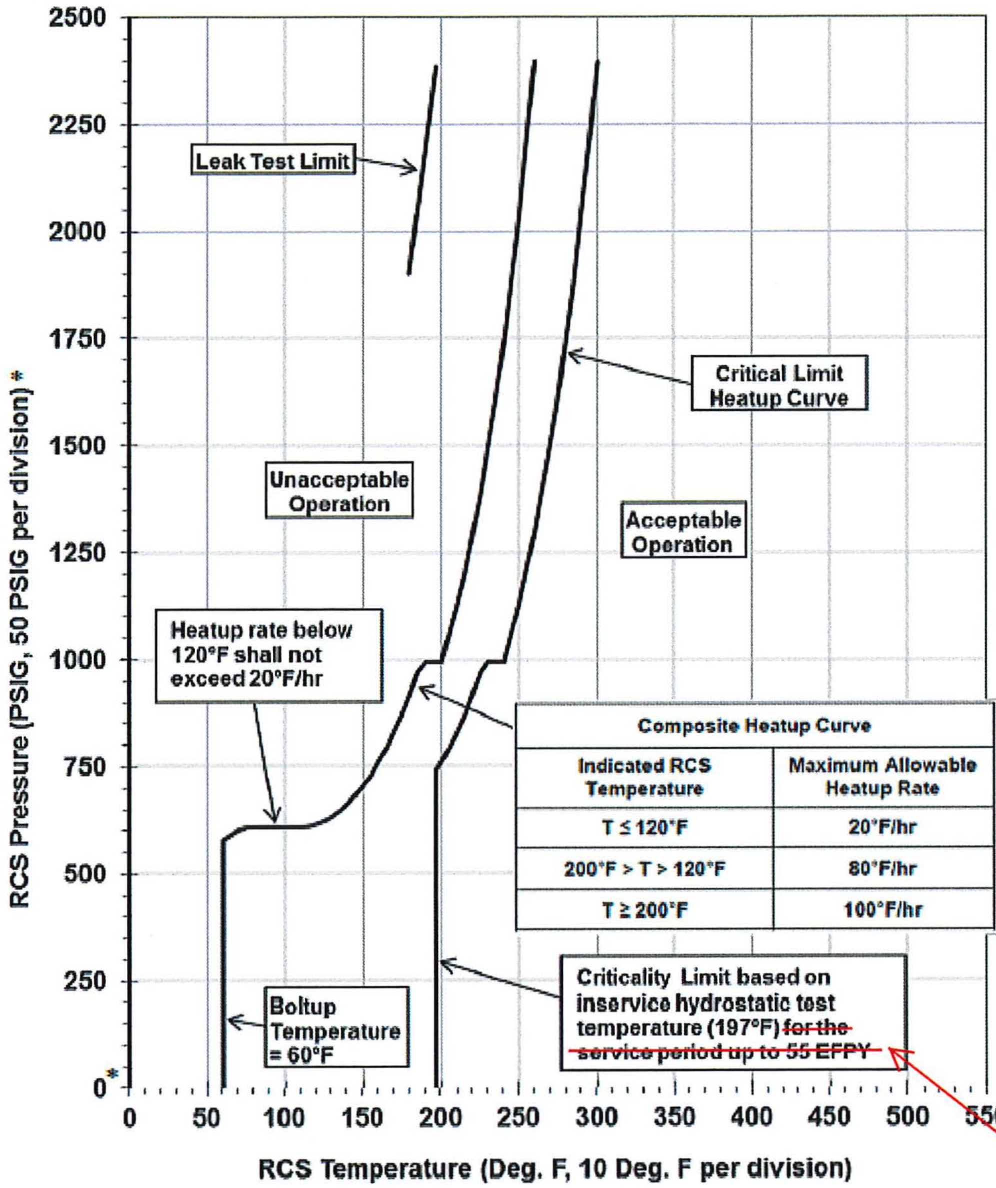
Remove

Remove

MATERIAL PROPERTY BASIS

LIMITING MATERIAL: Lower Shell Plate R1808-1 without using surveillance data, Position 1.1
LIMITING ART VALUES ~~AT 55 EFPY~~: 1/4T, 117°F (Axial Flaw)
3/4T, 105°F (Axial Flaw)

Curves ~~applicable for the first 55 EFPY and~~ contain margins for possible instrument errors



Remove

* Curve is Applicable for RCS Vacuum fill.

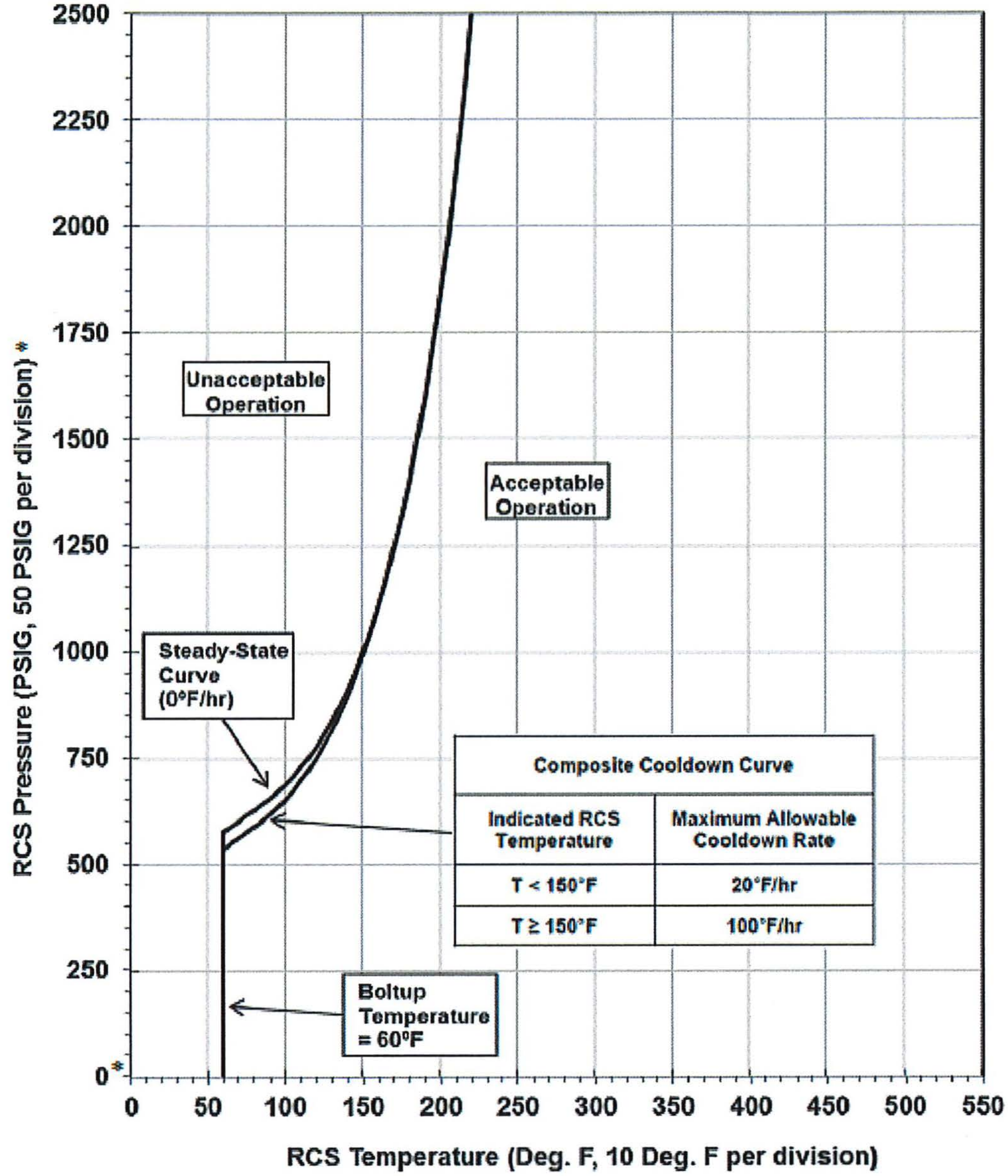
FIGURE 3.4-2
REACTOR COOLANT SYSTEM HEATUP LIMITATIONS ~~– APPLICABLE UP TO 55 EFPY~~

Remove

MATERIAL PROPERTY BASIS

LIMITING MATERIAL: Lower Shell Plate R1808-1 without using surveillance data, Position 1.1
LIMITING ART VALUES ~~AT 55 EFPY:~~ 1/4T, 117°F (Axial Flaw)
3/4T, 105°F (Axial Flaw)

Curves ~~applicable for the first 55 EFPY and~~ contain margins for possible instrument errors



* Curve is Applicable for RCS Vacuum fill.

Remove

FIGURE 3.4-3
REACTOR COOLANT SYSTEM COOLDOWN LIMITATIONS ~~– APPLICABLE UP TO 55 EFPY~~

Remove

~~VALID FOR THE FIRST 55 EFPY,~~ MAXIMUM SETPOINT ACCOUNTS FOR INSTRUMENT UNCERTAINTIES

$T \leq 75.0°F$, $P = 520.0$ PSIG;
$75.0°F < T \leq 125.0°F$, $P = 1.5*(T-75.0) + 520.0$ PSIG;
$125.0°F < T \leq 160.0°F$, $P = 3.71*(T-125.0) + 595.0$ PSIG;
$160.0°F < T \leq 350.0°F$, $P = 725.0$ PSIG *

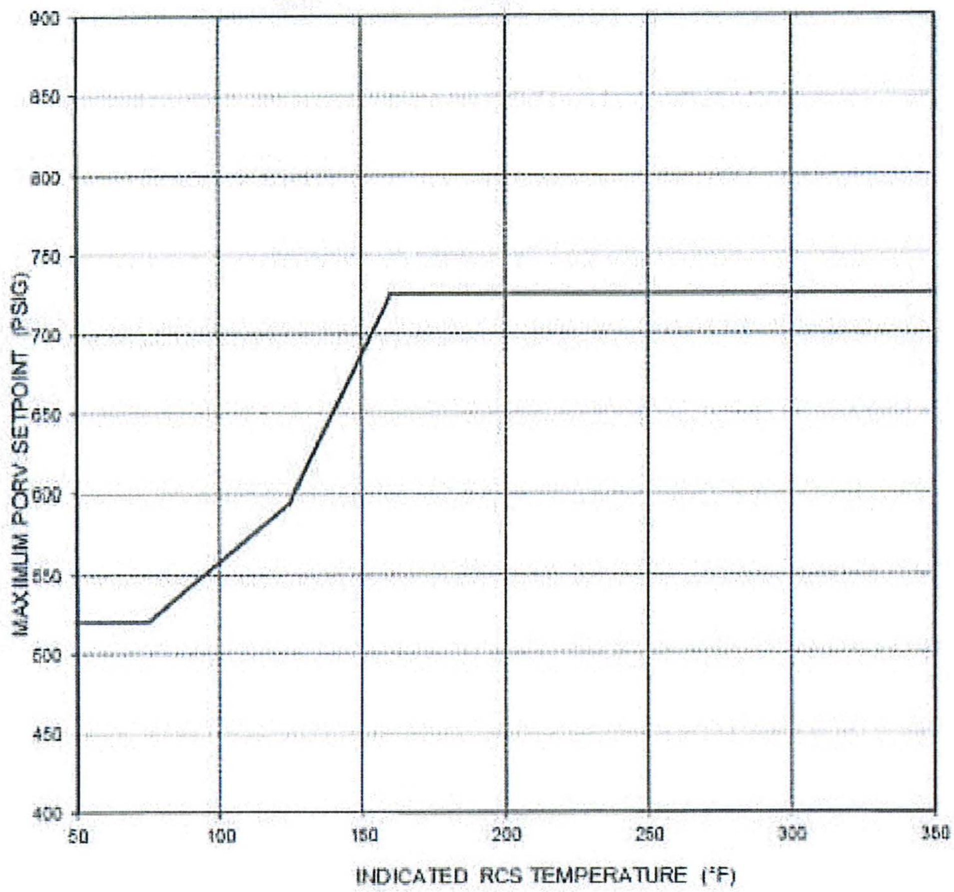


FIGURE 3.4-4 MAXIMUM ALLOWABLE PORV SETPOINTS FOR COLD OVERPRESSURE PROTECTION SYSTEM

*** Note that above the enable temperature the PORV setpoints will not restrict plant heatup and cooldown operations since COMS is not required to be armed at temperatures higher than 225°F. Hence the PORV setpoint values ramp up to the nominal setpoint value of 2385 psig is not shown.**

ATTACHMENT 2

PROPOSED TECHNICAL SPECIFICATION BASES CHANGES (MARK-UP)

(3 pages follow)

Updated fluence projections in the analysis of surveillance capsule X, Reference (7) resulted in the period of applicability being reduced from 55 EFPY to 52.6 EFPY, Reference (EC 296081)

3/4.4.9 PRESSURE/TEMPERATURE LIMITS (Continued)

were analyzed

The P/T limits have been established in accordance with the requirements of ASME Boiler and Pressure Vessel Code Section XI, Appendix G, and the additional requirements of 10CFR50 Appendix G, Reference (4). 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 55 effective full power years (EFPY). The technical justification and methodologies utilized in their development are documented generically in WCAP-14040-A, Revision 4, Reference (3), and specifically for Seabrook Unit 1 in WCAP-17441-NP, Reference (5), and LTR-AMLRS-11-50, Reference (8). 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 (6). 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.

four

V and X

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 (7), is available for ~~three~~ capsules (Capsules U, Y, ~~and V~~) 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 (7) 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} ~~but does not include~~ adjustments for possible errors in the pressure and temperature sensing instruments.

and

3/4.4.9 PRESSURE/TEMPERATURE LIMITS (Continued)

V and X

The results from the material surveillance program were evaluated according to ASTM E185. Capsules U, Y, ~~and V~~ were removed in accordance with the requirements of ASTM E185-82 and 10CFR50, Appendix H. The lead factor represents the relationship between the fast neutron flux density at the location of the capsule and the inner wall of the reactor vessel. Therefore, the results obtained from the surveillance specimens were used to predict future radiation damage to the reactor vessel material by using the lead factor and the withdrawal time of the capsule. The fluence values used to determine the CFs are the calculated fluence values at the surveillance capsule locations. The calculated fluence values were used for all cases. All calculated fluence values (capsule and projections) are documented in References (5) and (7). These fluences were calculated using the ENDF/B-VI scattering cross-section data set. The measured ΔRT_{NDT} values for the weld data were adjusted for chemistry using the ratio procedure given in Position 2.1 of Regulatory Guide 1.99, Revision 2. Since the ratio is equal to 1.0, the calculations are not affected by the ratio procedure.

3/4.4.9 PRESSURE/TEMPERATURE LIMITS (Continued)

HEATUP (Continued)

10 CFR Part 50, Appendix G, Reference (4), 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. However, per WCAP-17444-NP, Reference (9), Seabrook Unit 1 is justified for an exemption to these requirements. Therefore, these requirements are not contained in Figures 3.4-2 and 3.4-3.

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 1998 through 2000 Addenda.

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. Westinghouse WCAP-14040-A, Revision 4, "Methodology Used to Develop Cold Overpressure Mitigating Setpoints and RCS Heatup and Cooldown Limit Curves", dated May 2004.

4. 10 CFR Part 50, Appendix G, "Fracture Toughness Requirements", U.S. Nuclear Regulatory Commission, Federal Register, Volume 60, No. 243, dated December 19, 1995.

5. Westinghouse WCAP-17441-NP, Revision 0, "Seabrook Unit 1 Heatup and Cooldown Limit Curves for Normal Operation", dated October 2011.

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

7. Westinghouse WCAP-~~16526~~ 18067-NP, Revision 0, "Analysis of Capsule ~~V~~ X from ~~FPL~~ NextEra Energy-Seabrook Unit 1 Reactor Vessel Radiation Surveillance Program", dated March ~~2006~~ 2021