NEXTera® ENERGY SEABROOK

July 24, 2014
10 CFR 2.390
10 CFR 50.90
10 CFR 50.12

SBK-L-14102
Docket No. 50-443

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

Seabrook Station

License Amendment Request 14-04

Revised Reactor Coolant System Pressure - Temperature Limits Applicable for 55 Effective Full Power Years

Pursuant to 10 CFR 50.90, NextEra Energy Seabrook, LLC (NextEra) is submitting License Amendment Request (LAR) 14-04 to revise the Seabrook Station Technical Specification (TS). The proposed change revises the pressure-temperature (P/T) limits in TS 3.4.9.1, Reactor Coolant System Pressure-Temperature Limits, to be applicable to 55 effective full power years. The change also revises TS 3.4.9.3, Overpressure Protection Systems, by providing new overpressure protection setpoints and lowering the RCS temperature at which the TS is applicable.

The Enclosure to this letter provides NextEra's evaluation of the proposed change. Attachment 1 to the enclosure provides markups of the TS that show the proposed change, and Attachment 2 contains markups of the TS Bases. The proposed changes to the TS Bases are provided for information only and will be implemented in accordance with the TS Bases Control Program upon implementation of the requested amendment.

Attachment 3 contains WCAP-17441-NP, "Seabrook Unit 1 Heatup and Cooldown Limit Curves for Normal Operation." WCAP-17441-NP provides the methodology and results of the generation of heatup and cooldown P/T limit curves for normal operation of the Seabrook Unit 1 reactor vessel up to 55 EFPY. The PT Limit curves documented in WCAP-17441-NP were developed without the reactor vessel flange requirements of 10 CFR 50, Appendix G. Therefore, included in Attachment 4 is a request for an exemption from the requirements of 10 CFR 50, Appendix G.

Attachments 5 and 7 provide a nonproprietary version (WCAP-17444-NP) and a proprietary version (WCAP-17444-P), respectively, of the report "Reactor Vessel Closure Head/Vessel Flange Requirements Evaluation for Seabrook Unit 1." Attachment 7 contains information proprietary to Westinghouse Electric Company LLC, and is supported by an affidavit in Attachment 6 signed by Westinghouse, the owner of the information. The affidavit sets forth the basis on which the information may be withheld from public disclosure and addresses with specificity the considerations listed in paragraph (b)(4) of 10 CFR 2.390. Accordingly, NextEra requests that the information that is proprietary to Westinghouse (Attachment 7) be withheld from public disclosure in accordance with 10 CFR 2.390. Correspondence with respect to the copyright or proprietary aspects of the items listed above or the supporting Westinghouse affidavit should reference CAW-12-3366 and should be addressed to J. A. Gresham, Manager, Regulatory Compliance, Westinghouse Electric Company, Suite 428, 1000 Westinghouse Drive, Cranberry Township, Pennsylvania 16066.

As discussed in the evaluation, the proposed change does not involve a significant hazards consideration pursuant to 10 CFR 50.92, and there are no significant environmental impacts associated with the change.

No new commitments are made as a result of this change.

The Station Operation Review Committee has reviewed this LAR.

In accordance with 10 CFR 50.91, NextEra is notifying the State of New Hampshire of this LAR by transmitting a copy of this letter and enclosure to the designated State Official.

NextEra requests NRC review and approval of LAR 14-04 by February 1, 2016, and implementation within 30 days of the start of refueling outage 18, which is scheduled to commence on April 1, 2017.

Should you have any questions regarding this letter, please contact Mr. Michael Ossing, Licensing Manager, at (603) 773-7512.

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

Executed on July 24, 2014.

Sincerely,

Dean Curtland
Site Vice President
NextEra Energy Seabrook, LLC

Enclosure: Evaluation of the Proposed Change

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

Director Homeland Security and Emergency Management
New Hampshire Department of Safety
Division of Homeland Security and Emergency Management
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Enclosure

NextEra Energy Seabrook's Evaluation of the Proposed Change

Subject: Revised Reactor Coolant System Pressure - Temperature Limits Applicable for 55 Effective Full Power Years

1.0 SUMMARY DESCRIPTION

The proposed change revises Seabrook Technical Specification (TS) 3.4.9.1, Pressure / Temperature Limits, by providing new reactor coolant system (RCS) pressure and temperature (P/T) limits that are applicable to 55 effective full power years (EFPY). The change also revises TS 3.4.9.3, Overpressure Protection Systems, by providing new overpressure protection setpoints and lowering the RCS temperature at which the TS is applicable.

2.0 DETAILED DESCRIPTION

The proposed changes include the following:

1. TS 3.4.9.1, Pressure Temperature Limits
   - Figure 3.4-2, Reactor Coolant System Heatup Limitations - Applicable Up to 23.7 EFPY, is replaced with a new figure applicable to 55 EFPY.
   - Figure 3.4-3, Reactor Coolant System Cooldown Limitations - Applicable Up to 23.7 EFPY, is replaced with a new figure applicable to 55 EFPY.
2. TS 3.4.9.3, Overpressure Protections Systems
   - The TS limiting condition for operation is modified as shown below:

     a. In MODE 4 when the temperature of any RCS cold leg is less than or equal to ~~290~~ ***225*** °F; and in MODE 5 and MODE 6 with all Safety Injection pumps inoperable at least one of the following groups of two overpressure protection devices shall be OPERABLE when the RCS is not depressurized with an RCS vent area of greater than or equal to 1.58 square inches:
   - The TS applicability is modified as shown below:
     <u>APPLICABILITY:</u> MODE 4 when the temperature of any RCS cold leg is less than or equal to ~~290~~ ***225*** °F; MODE 5 and MODE 6 with the reactor vessel head on and the vessel head closure bolts not fully detensioned.
   - Figure 3.4-4, RCS Cold Overpressure Protection Setpoints, is replaced with a new Figure 3.4-4, Maximum Allowable PORV Setpoints for Cold Overpressure Protection System, which is valid for up to 55 EFPY.

3.0 TECHNICAL EVALUATION

3.1 <u>Pressure - Temperature (P/T) Limit Curves</u>

Background

The P/T limits are not derived from Design Basis Accident (DBA) analyses. They are prescribed during normal operation to avoid encountering pressure, temperature, and temperature rate of change conditions that might cause undetected flaws to propagate and cause non-ductile failure of the Reactor Coolant Pressure Boundary (RCPB), an unanalyzed condition. Although the P/T limits are not derived from any DBA, the P/T limits are acceptance limits since they preclude operation in an unanalyzed condition. All components of the RCS are designed to withstand effects of cyclic loads due to system pressure and temperature changes in accordance with 10 CFR 50, Appendix A, General Design Criterion 31, "Fracture prevention of reactor coolant pressure boundary," [Reference 1].

TS 3.4.9.1, "Pressure/Temperature Limits," contains the heatup and cooldown limit curves (Figures 3.4-2 and 3.4-3 respectively) that establish the limits for the pressure and temperature changes during RCS heatup and cooldown. The consequence of violating the TS 3.4.9.1 limits is that the RCS may have been operated under conditions that can result in brittle failure of the RCPB, possibly leading to a non-isolable leak or loss of coolant accident. The development of the P/T limits for the RCPB must consider not only the vessel shell material with the highest reference temperature but also other vessel materials with structural discontinuities, in particular the reactor vessel nozzle materials when the nozzles are positioned immediately above or below the active core height. Specifically, all ferritic components within the entire RCPB must be considered in the development of P/T limits and the effects of neutron radiation must be considered for any materials that are predicted to experience an end-of-license exposure greater than 1×10^{17} n/cm^2 (E>1 MeV). The present P/T limits consider all ferritic components within the entire RCPB. As documented in Reference 6, it was concluded that considering the reactor vessel inlet and outlet nozzles and the ferritic RCPB components not part of the reactor vessel, the P/T limit curves are bounding for the entire reactor vessel and meet the applicable requirements of the ASME Code, Section III, as required by 10CFR Part 50, Appendix G.

The P/T limit curves contained in TS 3.4.9.1 define an acceptable region for normal operation and establish operating limits that provide a margin to brittle failure of the reactor vessel and piping of the RCPB. The reactor vessel is the limiting RCPB component for establishing the P/T limit curves.

10 CFR 50, Appendix G, "Fracture Toughness Requirements," requires the establishment of P/T limits for specific material fracture toughness requirements of the RCPB materials. Reference 1 requires an adequate margin to brittle failure during normal operation, anticipated operational occurrences, and system hydrostatic tests. Reference 1 also requires the use of the American Society of Mechanical Engineers (ASME) Code, Section III, Appendix G [Reference 2].

The neutron embrittlement effect 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} of the vessel material is established periodically by removing and evaluating the irradiated reactor vessel material specimens, in accordance with ASTM E 185 [Reference 3] as required by Appendix H of 10 CFR 50, "Reactor Vessel Material Surveillance Program Requirements." 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 initial 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 operating P/T limit curves are adjusted, as necessary, based on the evaluation findings and the recommendations of Regulatory Guide 1.99, "Radiation Embrittlement of Reactor Vessel Materials," [Reference 4]. Regulatory Guide 1.99 provides the approved method for predicting radiation embrittlement. Regulatory Guide 1.99 is used for the calculation of 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 measured from the clad/base metal interface. The most limiting ART values are used in the generation of heatup and cooldown P/T limit curves for normal operation.

Evaluation

Attachment 3, WCAP-17441-NP, "Seabrook Unit 1 Heatup and Cooldown Limit Curves for Normal Operation," provides the methodology and results of the generation of heatup and cooldown P/T limit curves for normal operation of the Seabrook Unit 1 reactor vessel up to 55 EFPY. This document was previously provided in Reference 10 in support of the development of the present P/T limits and used to address all of the components comprising the RCPB. The heatup and cooldown P/T limit curves documented in WCAP-17441-NP were generated using the most limiting adjusted reference temperature values for Seabrook Unit 1 in accordance with the NRC-approved methodology documented in WCAP-14040-A, "Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves."

WCAP-14040-A utilizes the approved method for predicting radiation embrittlement contained in Regulatory Guide 1.99. As documented in Reference 6, the NRC staff reviewed WCAP-17441-NP and confirmed that the neutron fluence projections were calculated using the methods described in WCAP-14040-A. These neutron fluence calculation methods adhere to the guidance contained in RG 1.190 as documented in the NRC safety evaluation approving WCAP-14040-A. Based on this consideration, the NRC staff determined that the fluence calculations described in WCAP-17441-NP are acceptable.

NextEra's response to the NRC's request for additional information (RAI) [Reference 5] supporting Seabrook Amendment 135 [Reference 6] discussed that WCAP-14040-A did not consider the inlet/outlet nozzles and demonstrated that the P/T limit curves are bounding for the entire reactor pressure vessel (RPV) by the development of component specific P/T limit curves for the RPV inlet/outlet nozzles. Figures 1 and 2 provided in the RAI response compared the nozzle P/T limits to the beltline P/T limits derived in WCAP-15745. The component specific P/T limit curves for the inlet and outlet nozzles that supported the RAI response as well as the proposed 55 EFPY P/T limits were developed using 55 EFPY fluence. Utilizing the data provided in Table 6-4 of WCAP-17441-NP, the P/T limits derived in WCAP-17441-NP for the most limiting 100° F/hr cooldown case have been superimposed on those figures below and demonstrate that the beltline P/T limits derived for 55 EFPY are below and to the right of the nozzle curves and remain bounding.

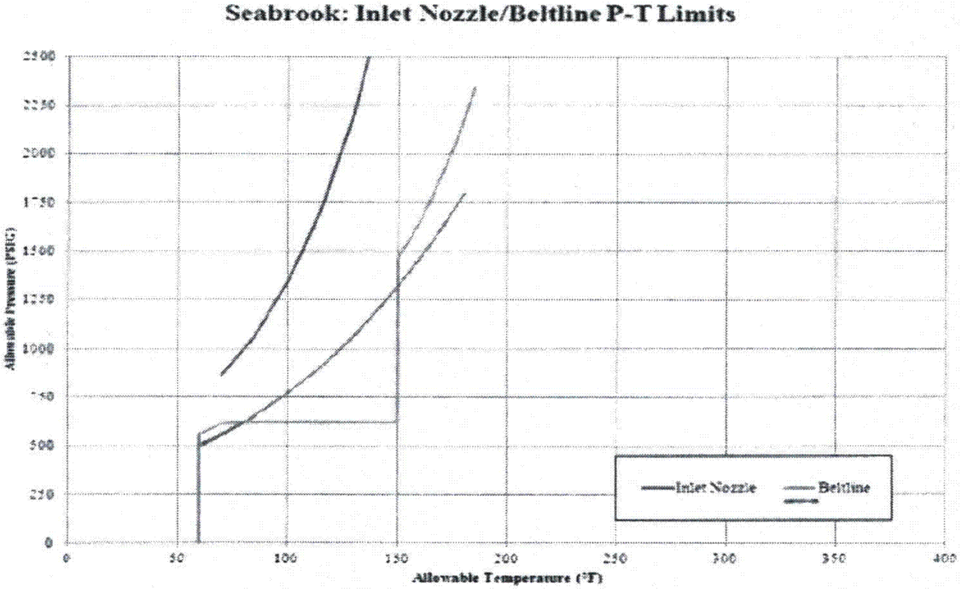


Figure 1: WCAP-17441 P-T Limits drawn on Comparison of Seabrook WCAP-15745 P-T Limits to Inlet Nozzle Limits [the superimposed red line is the 55 EFPY 100° F/hr curve]

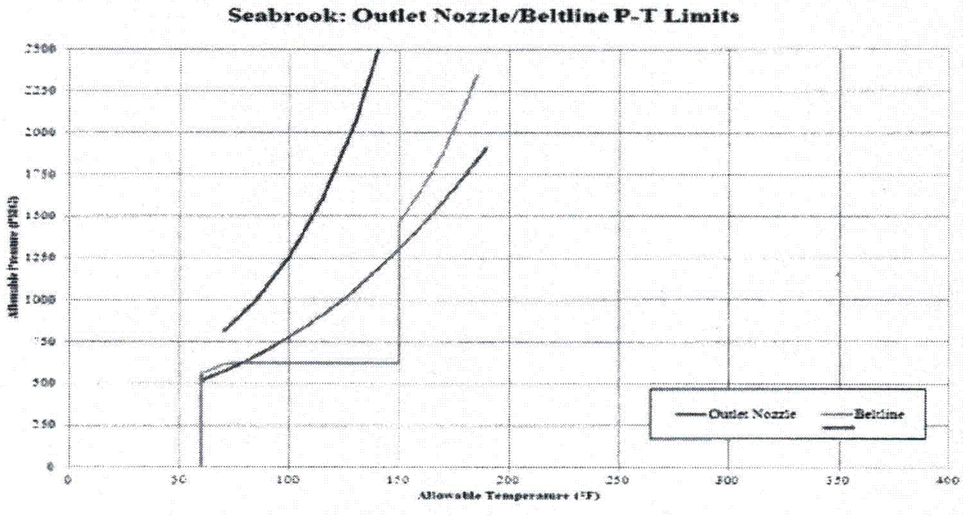


Figure 2: WCAP-17441 P-T Limits drawn on Comparison of Seabrook WCAP-15745 P-T Limits to Outlet Nozzle Limits [the superimposed red line is the 55 EFPY 100° F/hr curve]

The PT Limit curves documented in WCAP-17441-NP were developed without the reactor vessel flange requirements of 10 CFR 50, Appendix G. Therefore, an exemption to the requirements of 10 CFR 50, Appendix G is required. As provided for in Part (b) of 10 CFR 50.60 "Acceptance Criteria for Fracture Prevention Measures for Lightwater Nuclear Power Reactors for Normal Operation," Attachment 4 provides an exemption request, in accordance with 10 CFR 50.12, "Specific Exemptions."

The supporting justification for eliminating the requirements of 10 CFR 50, Appendix G for the closure head flange and vessel flange regions is contained in Attachment 7, WCAP-17444-P, "Reactor Vessel Closure Head/Vessel Flange Requirements Evaluation for Seabrook Unit 1."

Conclusion

The proposed 55 EFPY P/T limit curves were developed in accordance with the applicable requirements with the exception justified in WCAP-17444-P and in accordance with the NRC approved methodology contained in WCAP-14040-A. Therefore, the proposed 55 EFPY P/T limit curves will continue to provide an acceptable region for normal operation and operating limits that provide a margin to brittle failure of the reactor vessel and piping of the RCPB as required by 10 CFR 50, Appendix G.

The new P/T limit curves in TS 3.4.9.1 have been annotated indicating that the curve is applicable for RCS vacuum fill. With the combination of pressure and temperature maintained below and to the right of the limit lines in TS Figures 3.4-2 and 3.4-3, plant operation, including vacuum fill of the RCS, meets the limiting condition for operation of TS 3.4.9.1.

3.2 <u>RCS Cold Overpressure Protection Setpoints</u>

Background

TS 3.4.9.3, "Overpressure Protection Systems," contains the system requirements for overpressure protection during low temperature operation. The OPS provides the capability to mitigate the overpressure transients that may occur during cold shutdown, heatup, and cooldown operations to minimize the potential for challenging reactor vessel integrity when operating at or near reactor vessel ductility limits (i.e., 10 CFR 50, Appendix G limits). Two potential overpressure transients form the design basis and are considered in the analysis for low temperature overpressure protection; they are Mass Injection (MI) and Heat Injection (HI) transients. The design basis MI transient for Seabrook Unit 1 is specified as a result of starting a single centrifugal charging pump into a water solid RCS with letdown assumed to be isolated. The design basis HI transient for Seabrook Unit 1 is specified as the start of an idle RCP with all loops initially inactive and the SG secondary side a maximum of 50°F hotter than the RCS primary side.

In accordance with TS 3.4.9.3, "Overpressure Protection Systems," the pressurizer Power Operated Relief Valves (PORV) can be used to mitigate low temperature overpressure transients. The PORV setpoints are selected such that the peak pressure during the design basis MI and HI transients will not exceed the 10 CFR 50 Appendix G limits. The PORV setpoints are contained in TS Figure 3.4-4, "RCS Cold Overpressure Protection Setpoints." The current PORV setpoints for Seabrook Unit 1 are based on 10 CFR 50 Appendix G P/T limits valid through 23.7 EFPY.

The proposed change would update the 10 CFR 50 Appendix G P/T limits to be applicable to 55 EFPY. See Section 3.1 of this LAR for a description of the proposed change to the P/T limit curves. As discussed in Section 3.1 above, the revised P/T limit curves were developed without the 10 CFR 50 Appendix G reactor vessel flange requirements. Removal of the reactor vessel flange requirements provides higher operating margins for the PORV setpoints.

Based on the proposed P/T limit curve changes discussed above, new PORV setpoints were developed. As such, the proposed changes include a new TS Figure 3.4-4 applicable up to 55 EFPY to replace the current TS Figure 3.4-4 (applicable up to 23.7 EFPY). The proposed new TS Figure 3.4-4 is provided in Attachment 1.

Evaluation

The proposed PORV setpoints associated with the 55 EFPY P/T limits were developed in accordance with the guidance provided in NRC approved WCAP-14040-A, "Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves."

Consistent with the guidance provided in WCAP-14040-A, parametric analyses of the design basis MI and HI transients are performed for a range of PORV setpoints. The purpose of these parametric analyses is to generate the PORV setpoint pressure overshoot and undershoot data, which is used for the PORV setpoint determination. The PORV setpoint pressure overshoot and undershoot result from the PORV actuation to mitigate an increasing pressure transient. The PORV actuation results in the release of a volume of coolant through the valve that will cause the pressure increase to be slowed and reversed. The system pressure then decreases as the relief valve releases coolant until a reset pressure is reached where the PORV closes. The system pressure continues to decrease below the reset pressure as the valve recloses. The lower limit on the pressure during this transient is established based on an operational consideration for the reactor coolant pump number 1 seal to maintain the required differential pressure across the seal faces for proper seal operation. The upper pressure limit is based on protecting the 10CFR50 Appendix G limit and the PORV piping. As discussed in WCAP-14040-A, the 10CFR50 Appendix G limit and the reactor coolant pump number1 seal performance criteria create a pressure range from which the setpoints for both PORVs are selected.

In accordance with WCAP-14040-A, the peak RCS pressure resulting from the design basis MI and HI transients shall not exceed the minimum of the Appendix G limits and the PORV piping limit. Additionally, the minimum RCS pressure resulting from the design basis MI and HI transients should not drop below the reactor coolant pump number 1 seal differential pressure limit. However, if there is a conflict between satisfying the upper limits (i.e., the minimum of the 10 CFR 50 Appendix G limit and the PORV piping limit) and the lower limit (i.e., the reactor coolant pump number 1seal differential pressure limit), the upper pressure limits will take precedence.

Conclusion

The proposed change would revise the current TS Figure 3.4-4, to update the PORV setpoints consistent with the proposed 55 EFPY 10 CFR 50 Appendix G P/T limit curves discussed above in Section 3.1. The proposed PORV setpoints associated with the 55 EFPY P/T limits were determined in accordance with the NRC approved methodology contained in WCAP-14040-A. As such, the proposed PORV setpoints will continue to provide the required protection during low temperature operation (i.e., ensure 10 CFR 50 Appendix G limit is not exceeded) and will be applicable up to 55 EFPY consistent with the proposed new P/T limit curves.

3.3 Overpressure Protection Systems Effective Temperature

Background

TS 3.4.9.3, "Overpressure Protection Systems," contains the TS requirements for the OPS. The OPS limits RCS pressure at low temperatures so the integrity of the RCPB is not compromised by violating the pressure and temperature limits of 10 CFR 50, Appendix G, "Fracture Toughness Requirements." The reactor vessel is the limiting RCPB component for demonstrating such protection. The reactor vessel material is less tough at low temperatures than at normal operating temperature. RCS pressure, therefore, is maintained low at low temperatures and is increased only as temperature is increased. The OPS effective temperature is the limiting RCS cold leg temperature below which the reactor vessel may suffer damage from a cold overpressure event. Fracture mechanics analyses establish the OPS effective temperature. The OPS effective temperature is the temperature below which the system addressed in TS 3.4.9.3, "Overpressure Protection Systems," is required to be operable. As such, the OPS effective temperature is part of the Applicability of TS 3.4.9.3.

The OPS effective temperature is part of the TS requirements that establish the required protection of the 10 CFR 50 Appendix G P/T limits. As discussed in Section 3.1 above, the proposed changes include updating the 10 CFR 50 Appendix G P/T limits to be applicable to 55 EFPY. As such, the OPS effective temperature will also be revised consistent with the proposed change to the 10 CFR 50 Appendix G P/T limits.

Evaluation

The revised OPS effective temperature for Seabrook Unit 1 was calculated in accordance with ASME Code Case N-641, "Alternative Pressure- Temperature Relationship and Low Temperature Overpressure Protection System Requirements Section XI, Division 1" [Reference 7]. ASME Code Case N-641 presents alternative procedures for calculating pressure temperature relationships and OPS effective temperatures and allowable pressures. The procedures provided in Code Case N-641 take into account alternative fracture toughness properties, circumferential and axial reference flaws, and plant-specific OPS effective temperature calculations.

ASME Code Case N-641was first accepted by the NRC without conditions in Regulatory Guide (RG) 1.147, "Inservice Inspection Code Case Acceptability, ASME Section XI, Division 1, Revision 13" [Reference 8]. ASME Code Case N-641 continues to be identified as accepted without conditions in the current version of RG 1.147, Revision 16 [Reference 9]. RG 1.147, Revision 16, is identified as an approved standard in 10 CFR 50.55a(b), "Standards approved for Incorporation by Reference."

The calculated OPS effective temperature applicable for the proposed 55 EFPY P/T limits is 200°F.

The revised OPS effective temperature value that will be specified in TS 3.4.9.3, "Overpressure Protection Systems," includes a temperature uncertainty of 21.3°F (including indication errors). Accounting for instrument uncertainty ensures that the OPS is operable at the temperatures where it is required, consistent with the methodology used to calculate the temperature. Thus, the calculated OPS effective temperature plus instrument uncertainty is 221.3°F (200.0°F + 21.3°F = 221.3°F). Conservatively rounding this number up to the nearest 5°F provides a final effective temperature of 225°F for inclusion in TS 3.4.9.3, "Overpressure Protection Systems," to replace the current OPS effective temperature of 290°F.

Conclusion

The proposed change would revise the current OPS effective temperature value specified in TS 3.4.9.3 to be consistent with the proposed 55 EFPY 10CFR50 Appendix G P/T limit curves discussed above in Section 3.1. The proposed OPS effective temperature associated with the 55 EFPY P-T limits was determined in accordance with an approved ASME methodology (i.e., ASME Code Case N-641). As such, the proposed OPS effective temperature will continue to ensure the OPS is operable when required to protect the 10CFR50 Appendix G limit during low temperature operation.

4.0 REGULATORY EVALUATION

4.1 Applicable Regulatory Requirements/Criteria

- 10 CFR 50 Appendix A, "General Design Criteria for Nuclear Power Plants," Criterion 31, "Fracture prevention of reactor coolant pressure boundary," which states:

  "The reactor coolant pressure boundary shall be designed with sufficient margin to assure that when stressed under operating, maintenance, testing, and postulated accident conditions (1) the boundary behaves in a nonbrittle manner and (2) the probability of rapidly propagating fracture is minimized. The design shall reflect consideration of service temperatures and other conditions of the boundary material under operating, maintenance, testing, and postulated accident conditions and the uncertainties in determining (1) material properties, (2) the effects of irradiation on material properties, (3) residual, steady state and transient stresses, and (4) size of flaws."

- 10 CFR 50, Appendix G, "Fracture Toughness Requirements," which states in part:

  "This appendix specifies fracture toughness requirements for ferritic materials of pressure-retaining components of the reactor coolant pressure boundary of light water nuclear power reactors to provide adequate margins of safety during any condition of normal operation, including anticipated operational occurrences and system hydrostatic tests, to which the pressure boundary may be subjected over its service lifetime.

- Appendix H of 10 CFR 50, "Reactor Vessel Material Surveillance Program Requirements," which states in part:

  "The purpose of the material surveillance program required by this appendix is to monitor changes in the fracture toughness properties of ferritic materials in the reactor vessel beltline region of light water nuclear power reactors which result from exposure of these materials to neutron irradiation and the thermal environment. Under the program, fracture toughness test data are obtained from material specimens exposed in surveillance capsules, which are withdrawn periodically from the reactor vessel. These data will be used as described in Section IV of Appendix G to Part 50.

  ASTM E 185-73, "Standard Recommended Practice for Surveillance Tests for Nuclear Reactor Vessels"; ASTM E 185-79, "Standard Practice for Conducting Surveillance Tests for Light-Water Cooled Nuclear Power Reactor Vessels", and ASTM E 185-82, "Standard Practice for Conducting Surveillance Tests for Light-Water Cooled Nuclear Power Reactor Vessels"; which are referenced in the following paragraphs, have been approved for incorporation by reference by the Director of the Federal Register."

- General Design Criterion 31, *Fracture prevention of reactor coolant pressure boundary, requires that t*he reactor coolant pressure boundary be designed with sufficient margin to assure that when stressed under operating, maintenance, testing, and postulated accident conditions (1) the boundary behaves in a nonbrittle manner and (2) the probability of rapidly propagating fracture is minimized. The design shall reflect consideration of service temperatures and other conditions of the boundary material under operating, maintenance, testing, and postulated accident conditions and the uncertainties in determining (1) material properties, (2) the effects of irradiation on material properties, (3) residual, steady state and transient stresses, and (4) size of flaws.

- Regulatory Guide 1.99, Rev. 2,"Radiation Embrittlement of Reactor Vessel Materials, "May 1988, describes general procedures acceptable to the NRC staff for calculating the effects of neutron radiation embrittlement of the low-alloy steels currently used for light-water-cooled reactor vessels

- 10 CFR 50.60 "Acceptance criteria for fracture prevention measures for lightwater nuclear power reactors for normal operation.," which states:

  "(a) Except as provided in paragraph (b) of this section, all light-water nuclear power reactors, other than reactor facilities for which the certifications required under § 50.82(a)(1) have been submitted, must meet the fracture toughness and material surveillance program requirements for the reactor coolant pressure boundary set forth in appendices G and H to this part.

  (b) Proposed alternatives to the described requirements in Appendices G and H of this part or portions thereof may be used when an exemption is granted by the Commission under § 50.12."

- 10 CFR 50.12, "Specific Exemptions," states in part: " The Commission may, upon application by any interested person or upon its own initiative, grant exemptions from the requirements of the regulations of this part."

- Regulatory Guide 1.147, "Inservice Inspection Code Case Acceptability, ASME Section XI, Division 1," Revision 16, October 2010, which states in part:

  "General Design Criterion (GDC) 1, "Quality Standards and Records," of Appendix A, "General Design Criteria for Nuclear Power Plants," to Title 10, Part 50, of the *Code of Federal Regulations* (10 CFR Part 50), "Domestic Licensing of Production and Utilization Facilities" (Ref. 1), requires, in part, that structures, systems, and components important to safety be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety functions to be performed. Where generally recognized codes and standards are used, Criterion 1 requires that they be identified and evaluated to determine their applicability, adequacy, and sufficiency and be supplemented or modified as necessary to ensure a quality product in keeping with the required safety function."

The changes proposed in this request will continue to meet the above regulatory requirements as described in this LAR.

4.2 Precedent

The NRC staff has approved other requests for P/T limits applicable to end of license and exemption requests to the flange requirements of 10 CFR 50, Appendix G in the following NRC approval letters:

- Oconee Nuclear Station, Units 1, 2 and 3, Issuance of Amendments Regarding Revised Pressure - Temperature Limits (TAC Nos. MF0763, MF0764, and MF0765), February 27, 2014 [ML14041A093]
- Three Mile Island Nuclear Station, Unit 1 – Issuance of Amendment Re: Revision to the Pressure and Temperature Limit Curves and the Low Temperature Overpressure Protection Limits (TAC No. MF0424), December 13, 2013 [ML13325A023]
- Cooper Nuclear Station – Issuance of Amendment Re: Revisions to Technical Specification 3.4.9, "RCS Pressure and Temperature (P/T) Limits," for 32 Effective Full Power Years (TAC No. ME7324), February 22, 2013 [ML13032A526]
- Byron Station, Unit Nos. 1 and 2 and Braidwood Station Unit Nos. 1 and 2-Exemption from the Requirements of 10 CFR Part 50, Appendix G (TAC Nos. MC8697, MC8698, MC8699, and MC8700), November 22, 2006 [ML061890003]
- Vogtle Electric Generating Plant, Units 1 and 2 - Issuance of Exemption and Amendments Re: Request to Revise Technical Specifications and Pressure Temperature Limits Report and Relocate the Cold Overpressure Protection System (COPS) Arming Temperature (TAC Nos. MC2225, MC2226, MC2227, MC2228, MC3090, and MC3091), March 28, 2005 [ML050690216]
- Sequoyah Nuclear Plant, Units 1 and 2, Exemption from the Requirements of 10 CFR Part 50, Appendix G (TAC Nos. MB7321 and MB7322), July 7, 2004 [ML041940552]

4.3 Significant Hazards Consideration

No Significant Hazards Consideration

The proposed change revises Seabrook Technical Specification (TS) 3.4.9.1, Pressure / Temperature Limits, by providing new reactor coolant system (RCS) pressure and temperature (P/T) limits that are applicable to 55 effective full power years (EFPY). The change also revises TS 3.4.9.3, Overpressure Protection Systems, by providing new overpressure protection setpoints and lowering the RCS temperature at which the TS is applicable.

In accordance with 10 CFR 50.92, NextEra Energy Seabrook has concluded that the proposed change does not involve a significant hazards consideration (SHC). The basis for the conclusion that the proposed change does not involve a SHC is as follows:

1. *Does the proposed change involve a significant increase in the probability or consequences of an accident previously evaluated?*

   No. The proposed changes to the Technical Specifications (TS) do not impact the physical function of plant structures, systems, or components (SSCs) or the manner in which SSCs perform their design function. Operation in accordance with the proposed TS will ensure that all analyzed accidents will continue to be mitigated by the SSCs as previously analyzed. The proposed changes do not alter or prevent the ability of operable SSCs to perform their intended function to mitigate the consequences of an initiating event within assumed acceptance limits. The proposed changes neither adversely affect accident initiators or precursors, nor alter design assumptions.

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

2. *Does the proposed change create the possibility of a new or different kind of accident from any previously evaluated?*

   No. The proposed changes do not involve a physical alteration of the plant (i.e., no new or different type of equipment will be installed), create new failure modes for existing equipment, or create any new limiting single failures. The changes to the pressure - temperature limits, power operated relief valve setpoints, and the over pressure protection system effective temperature will continue to ensure that appropriate fracture toughness margins are maintained to protect against reactor vessel failure, during both normal and low temperature operation. The proposed changes are consistent with the applicable NRC approved methodologies (i.e., WCAP-14040, Rev. 4 and ASME Code Case N-641). Plant operation will not be altered, and all safety functions will continue to perform as previously assumed in accident analyses.

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

3. *Does the proposed change involve a significant reduction in the margin of safety?*

   No. Margin of safety is associated with confidence in the ability of the fission product barriers (i.e., fuel cladding, reactor coolant system pressure boundary, and containment structure) to limit the level of radiation dose to the public. The proposed changes will not adversely affect the operation of plant equipment or the function of any equipment assumed in the accident analysis. The proposed changes were developed using NRC approved methodologies and will continue to ensure an acceptable margin of safety is maintained. The safety analysis acceptance criteria are not affected by this change. The proposed changes will not result in plant operation in a configuration outside the design basis. The proposed changes do not adversely affect systems that respond to safely shutdown the plant and to maintain the plant in a safe shutdown condition.

Therefore, the proposed changes do not involve a significant reduction in a margin of safety.

Based on the above, NextEra concludes that the proposed amendment does not involve a significant hazards consideration under the standards set forth in 10 CFR 50.92(b), and, accordingly, a finding of "no significant hazards consideration" is justified.

4.4 Conclusions

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

NextEra has evaluated the proposed amendment for environmental considerations. The 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 significant increase in the amounts of any effluent 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 for in 10 CFR 51.22(c)(9). Therefore, pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment needs to be prepared in connection with the proposed amendment.

6.0 REFERENCES

1. 10 CFR 50, Appendix A, General Design Criterion 31, Fracture prevention of reactor coolant pressure boundary

2. Appendix G to the 1998 through the 2000 Addenda Edition of the ASME Boiler and Pressure Vessel (B&PV) Code, Section XI, Division 1, “Fracture Toughness Criteria for Protection Against Failure”

3. ASTM E 185-73, "Standard Recommended Practice for Surveillance Tests for Nuclear Reactor Vessels"; ASTM E 185-79, "Standard Practice for Conducting Surveillance Tests for Light-Water Cooled Nuclear Power Reactor Vessels"; and ASTM E 185-82, "Standard Practice for Conducting Surveillance Tests for Light-Water Cooled Nuclear Power Reactor Vessels," (as modified by Appendix H to 10 CFR Part 50, “Reactor Vessel Material Surveillance Program Requirements”)

4. Regulatory Guide 1.99, Radiation Embrittlement of Reactor Vessel Materials, Revision 2, May 1988

5. NextEra Energy Seabrook, LLC letter SBK-L-12279, “Response to Request for Additional Information Regarding License Amendment Request to change applicability of Technical Specification Pressure-Temperature Limits,” January 9, 2013 [ML13014A624]

6. NRC letter “Seabrook Station Unit No. 1 - Issuance of Amendment Re: Revision to the Applicability of the Reactor Coolant System Pressure - Temperature Limits and the Cold Overpressure Protection Setpoint (TAC No. ME7645),” April 15, 2013 [ML120820510]

7. ASME Code Case N-641, “Alternative Pressure-Temperature Relationship and Low Temperature Overpressure Protection System Requirements Section XI, Division 1,” January 17, 2000

8. Regulatory Guide 1.147, Inservice Inspection Code Case Acceptability, ASME Section XI, Division 1, Revision 13, June 2003

9. Regulatory Guide 1.147, Inservice Inspection Code Case Acceptability, ASME Section XI, Division 1, Revision 16, October 2010

10. NextEra Energy Seabrook, LLC letter 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,” December 3, 2012 [ML12341A095]

Attachment 1

Markup of the Technical Specifications

Replace this Figure and information with the Figure and information on the following page

MATERIAL PR

Limiting materia

Limiting ART values at 23.7 EFPY 1/4T, 109°F

3/4T, 88°F

Curves applicable for the first 23.7 EFPY and contain margins of 20°F and 100 psig for possible instrument errors

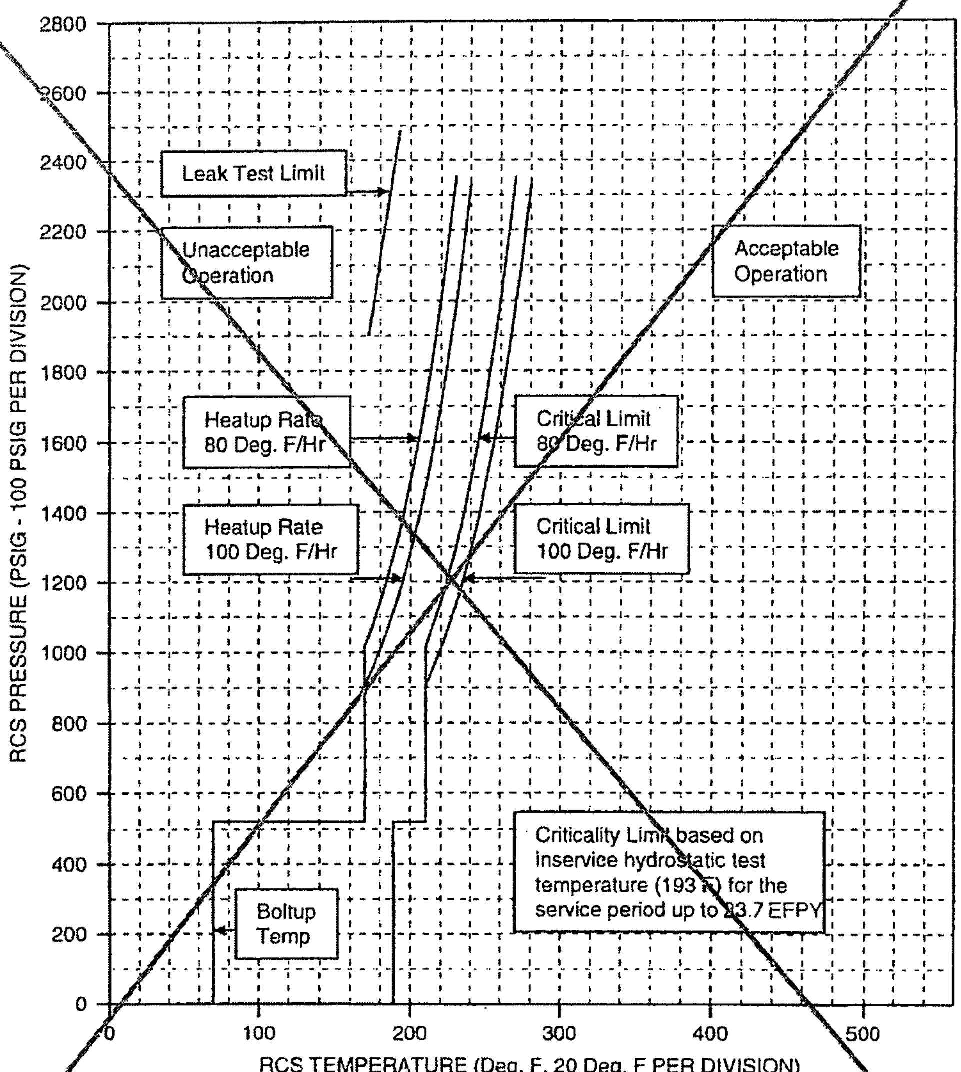


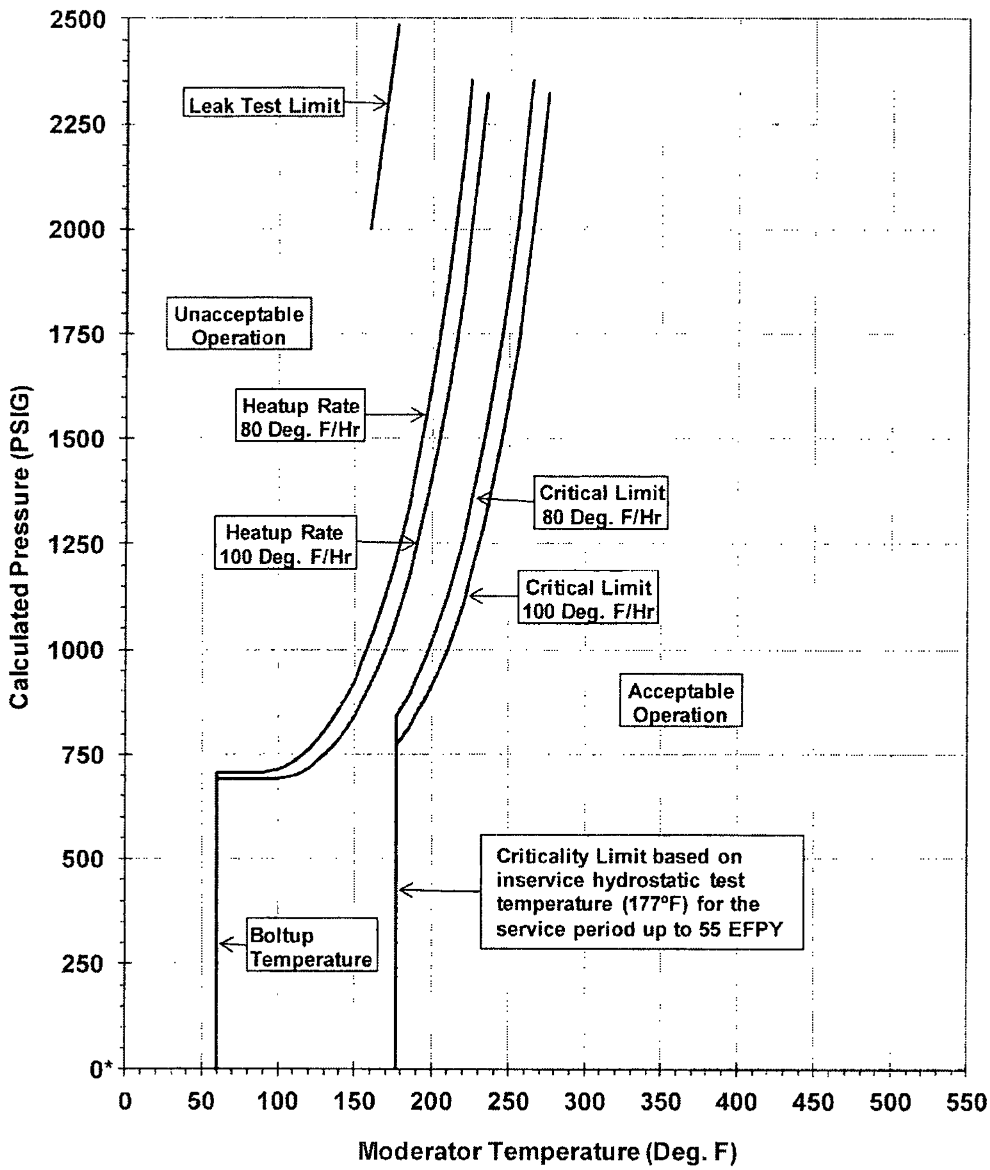
FIGURE 3.4-2

REACTOR COOLANT SYSTEM HEATUP LIMITATIONS - APPLICABLE UP TO 23.7 EFPY

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 do not contain margins for possible instrument errors



* Curve is applicable during RCS vacuum fill

FIGURE 3.4-2

REACTOR COOLANT SYSTEM HEATUP LIMITATIONS – APPLICABLE UP TO 55 EFPY

Replace this Figure and information with the Figure and information on the following page.

MATERIAL PRO

Limiting material

Limiting ART values at 23.7 EFPY:1/4T, 109°F
3/4T, 88°F

Curves applicable for the first 23.7 EFPY and contain margins of 20°F and 100 psig for possible instrument errors

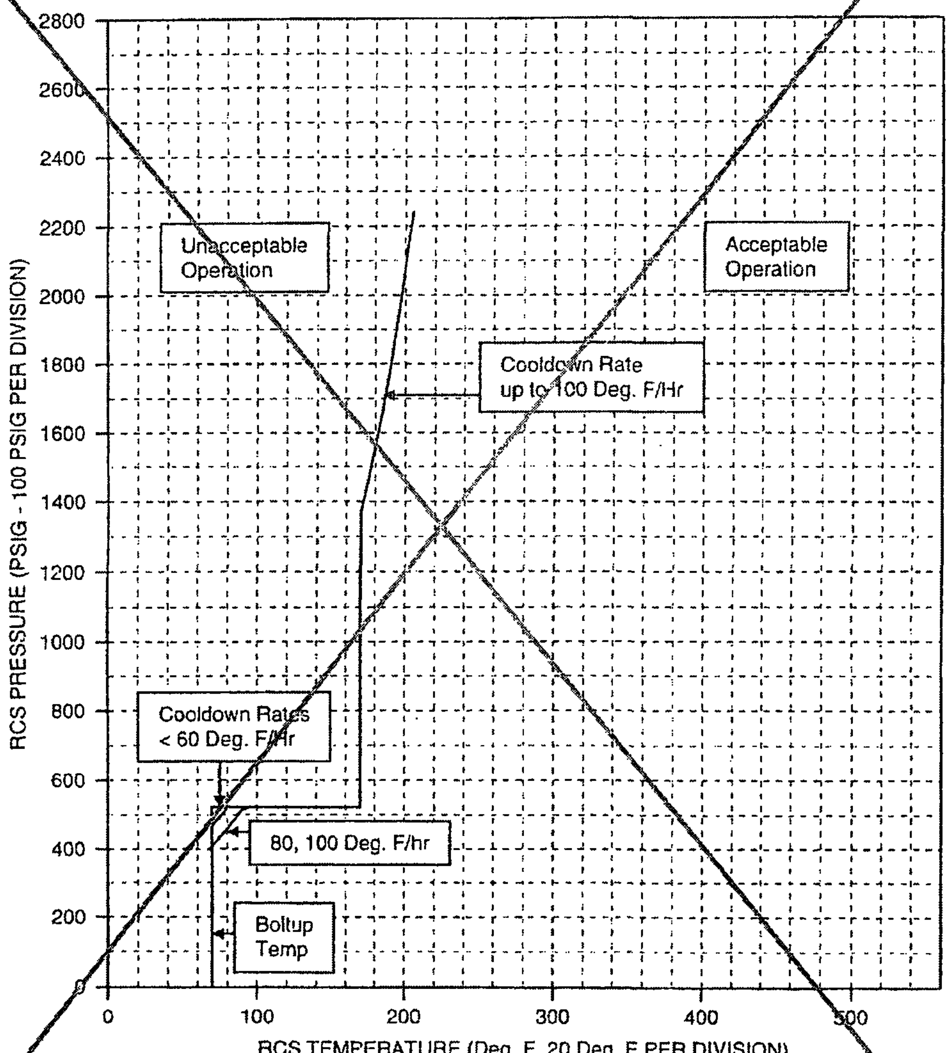


FIGURE 3.4-3
REACTOR COOLANT SYSTEM COOLDOWN LIMITATIONS - APPLICABLE UP TO 23.7 EFPY

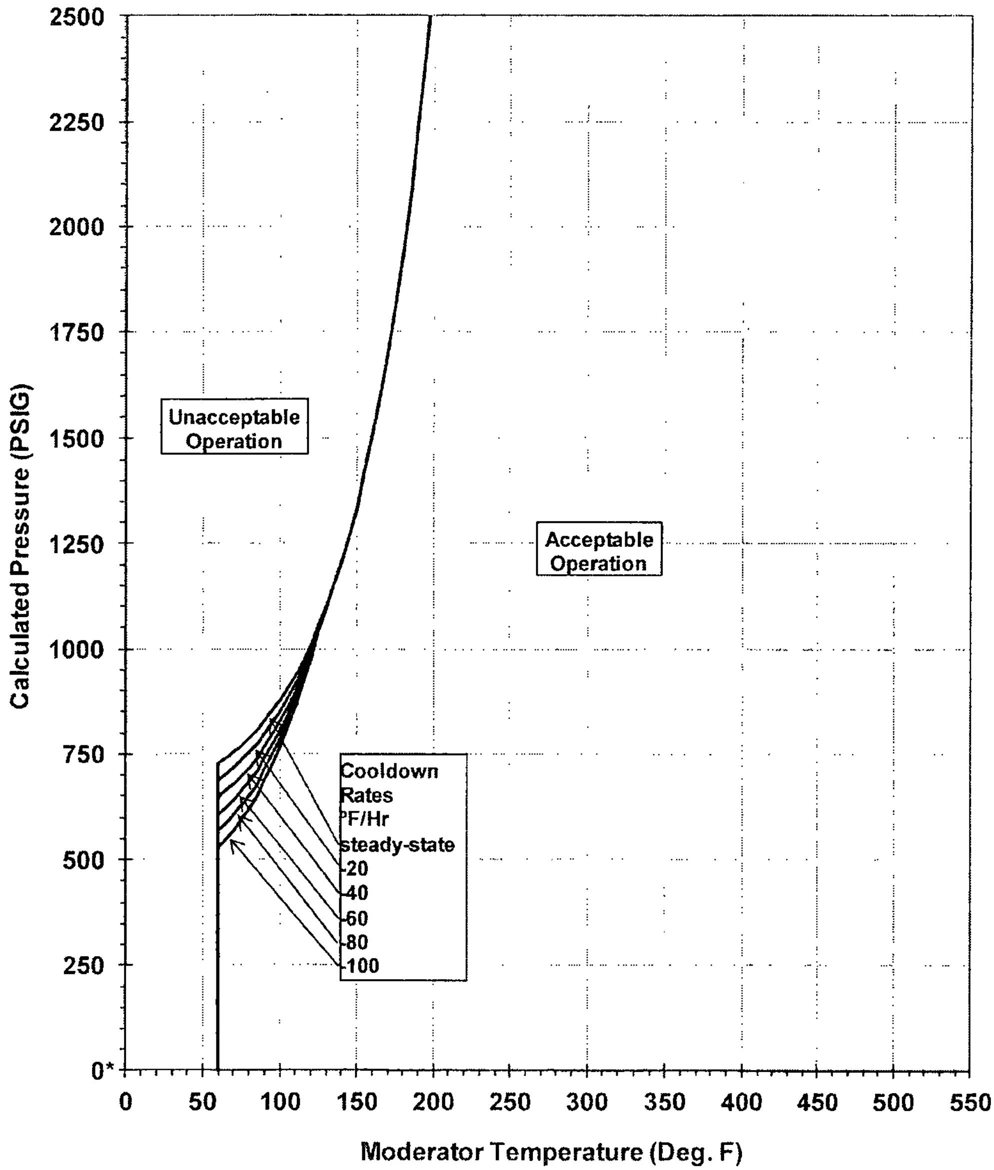
MATERIAL PROPERTY BASIS

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 do not contain margins for possible instrument errors



* Curve is applicable for RCS vacuum fill

FIGURE 3.4-3

REACTOR COOLANT SYSTEM COOLDOWN LIMITATIONS – APPLICABLE UP TO 55 EFPY

OVERPRESSURE PROTECTION SYSTEMS

LIMITING CONDITION FOR OPERATION

3.4.9.3 The following Overpressure Protection Systems shall be OPERABLE:

a. In MODE 4 ~~when the~~ temperatu 225 RCS cold leg is less than or equal to ~~290~~°F; and in MODE 5 and MODE 6 with all Safety Injection pumps inoperable at least one of the following groups of two overpressure protection devices shall be OPERABLE when the RCS is not depressurized with an RCS vent area of greater than or equal to 1.58 square inches:

   1) Two residual heat removal (RHR) suction relief valves each with a setpoint of 450 psig +0, -3 %; or

   2) Two power-operated relief valves (PORVs) with lift setpoints that vary with RCS temperature which do not exceed the limit established in Figure 3.4-4, or

   3) One RHR suction relief valve and one PORV with setpoints as required above.

b. In MODE 5 and MODE 6 with all Safety Injection pumps except one inoperable:

   1) The Reactor Coolant System (RCS) depressurized with an RCS vent area equal to or greater than 18 square inches, or

   2) The RCS in a reduced inventory condition*.

225

APPLICABILITY: MODE 4 when the temperature of any RCS cold leg is less than or equal to ~~290~~°F; MODE 5 and MODE 6 with the reactor vessel head on and the vessel head closure bolts not fully detensioned.

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*A reduced inventory condition exists whenever reactor vessel (RV) water level is lower than 36 inches below the RV flange.

Replace this figure and information with the Figure and information on the following page

VALID FOR THE FIRST 23.7 EFPY, SETPOINT CONTAINS MARGIN OF 50°F FOR TRANSIENT EFFECTS

$T \leq 200.0°F$, $P = 561.0$ PSIG;
$200.0°F < T \leq 230.5°F$, $P = 12.1*(T-200.0) + 926.0$ PSIG;
$230.5°F < T \leq 255.0°F$, $P = 23.15*(T-230.5) + 1295.05$ PSIG;
$T > 255.0°F$, $P = 34.5*(T-255.0) + 1862.225$ PSIG

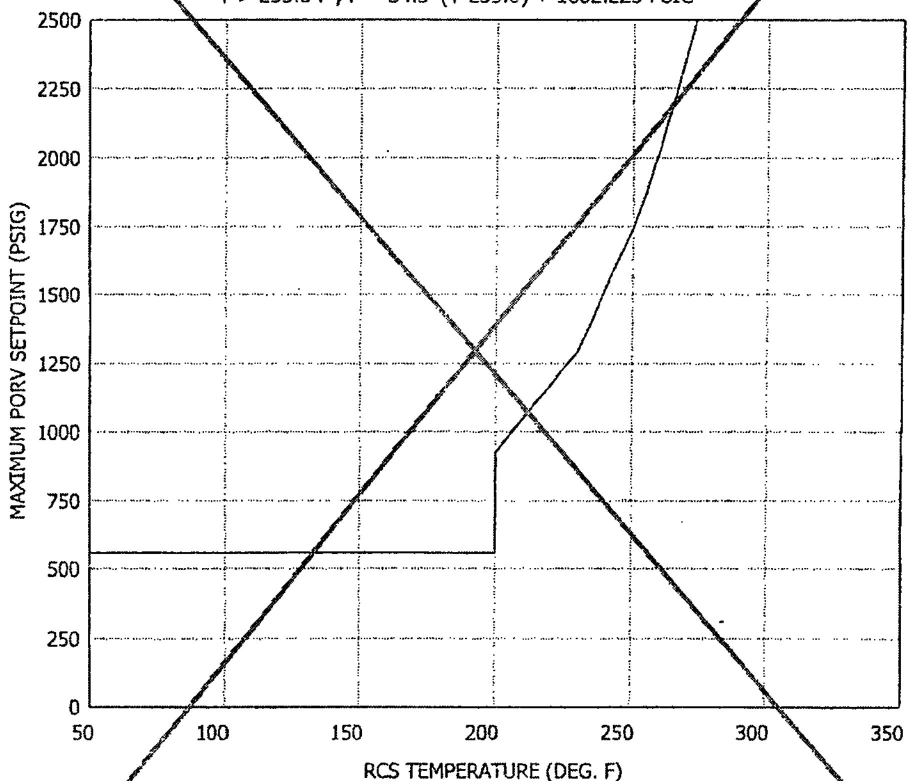

FIGURE 3.4-4 RCS COLD OVERPRESSURE PROTECTION SETPOINTS

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

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

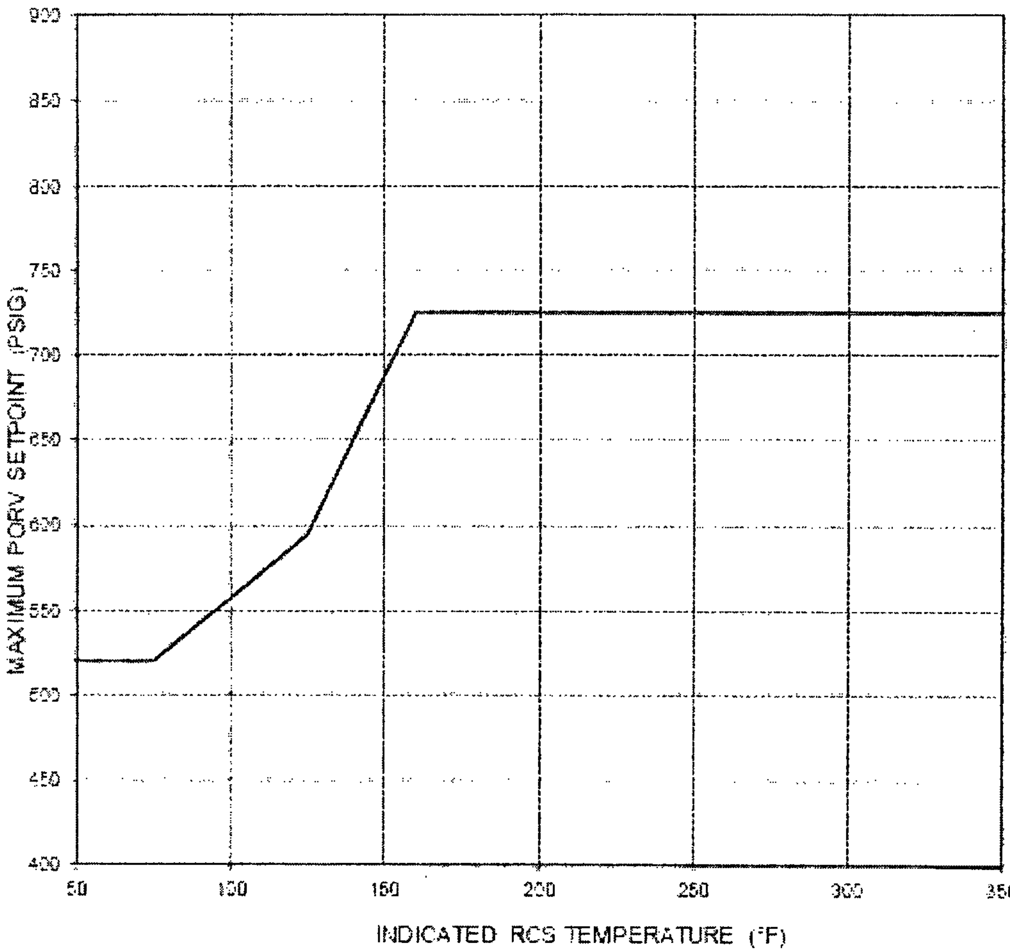

FIGURE 3.4-4 MAXIMUM ALLOWABLE PORV SETPOINTS FOR COLD OVERPRESSURE PROTECTION SYSTEM

*** 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

Markup of the Technical Specifications Bases

3/4.4.9 PRESSURE/TEMPERATURE LIMITS

The temperature and pressure changes during heatup and cooldown are limited to be consistent with the requirements given in the ASME Boiler and Pressure Vessel Code, Section XI, Appendix G, Reference (1):

1. The reactor coolant temperature and pressure and system heatup and cooldown rates (with the exception of the pressurizer) shall be limited in accordance with Figures 3.4-2 and 3.4-3 for the service period specified thereon:

   a. Allowable combinations of pressure and temperature for specific temperature change rates are below and to the right of the limit lines shown. Limit lines for cooldown rates between those presented may be obtained by interpolation; and

   b. Figures 3.4-2 and 3.4-3 define limits to assure prevention of non-ductile failure only. For normal operation, other inherent plant characteristics, e.g., pump heat addition and pressurizer heater capacity, may limit the heatup and cooldown rates that can be achieved over certain pressure-temperature ranges.

2. These limit lines shall be calculated periodically using methods provided below,

3. The secondary side of the steam generator must not be pressurized above 200 psig if the temperature of the steam generator is below 70°F,

4. The pressurizer heatup and cooldown rates shall not exceed 100°F/h and 200°F/h, respectively. The spray shall not be used if the temperature difference between the pressurizer and the spray fluid is greater than 320°F, and

5. System preservice hydrotests and inservice leak and hydrotests shall be performed at pressures in accordance with the requirements of ASME Boiler and Pressure Vessel Code, Section XI.

Operation within the limits of the appropriate heatup and cooldown curves assures the integrity of the reactor vessel's ferritic material against fracture induced by combined thermal and pressure stresses. As the reactor vessel is subjected to increasing fluence, the toughness of the limiting beltline region material continues to diminish, and consequently, even more restrictive pressure/temperature (P/T) limits must be maintained. Each P/T limit curve defines an acceptable region for normal operation during heatup or cooldown maneuvering as pressure and temperature indications are monitored to ensure that operation is within the allowable region. A heatup or cooldown is defined as a temperature change of greater than or equal to 10°F in any one-hour period.

With the combination of pressure and temperature maintained below and to the right of the limit lines in TS Figures 3.4-2 and 3.4-3, plant Operation, including vacuum fill of the RCS, meets the LCO of TS 3.4.9.1. The limit curves are applicable during RCS vacuum fill.

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 (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 ~~23.7~~ 55 effective full power years (EFPY). The technical justification and methodologies utilized in their development are documented ~~in WCAP-15745~~ 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.

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 ~~two~~ three capsules (Capsules ~~U and Y~~ 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} ~~as well as~~ but does not include adjustments for possible errors in the pressure and temperature sensing instruments.

3/4.4.9 PRESSURE/TEMPERATURE LIMITS (Continued)

82

U, Y and V

The results from the ~~material~~ surveillance program were evaluated according to ASTM E185. Capsules ~~U and Y~~ were removed in accordance with the requirements of ASTM E185-~~73~~ 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 Reference (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.

s (5) and

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)

The ASME approach for calculating the allowable limit curves for various heatup and cooldown rates specifies that the total stress intensity factor, K_I, for the combined thermal and pressure stresses at any time during heatup or cooldown cannot be greater than the reference stress intensity factor, K_{Ic}, for the metal temperature at that time. K_{Ic} is obtained from the reference fracture toughness curve, defined in ~~Code Case N-641, Reference (2).~~ The K_{Ic} curve is given by the following equation:

$$K_{Ic} = 33.2 + 20.734 * e^{[0.02(T - RT_{NDT})]} \qquad (1)$$

where,

K_{Ic} = reference stress intensity factor as a function of the metal temperature T and the metal reference nil-ductility temperature RT_{NDT}

This K_{Ic} curve is based on the lower bound of static critical K_I values measured as a function of temperature on specimens of SA-533 Grade B Class 1, SA-508-1, SA-508-2, and SA-508-3 steel.

The governing equation for the heatup-cooldown analysis is defined in Appendix G of the ASME Code as follows:

$$C^* K_{Im} + K_{It} < K_{Ic} \qquad (2)$$

where,

K_{Im} = stress intensity factor caused by membrane (pressure) stress

K_{It} = stress intensity factor caused by the thermal gradients

K_{Ic} = function of temperature relative to the RT_{NDT} of the material

C = 2.0 for Level A and Level B service limits

C = 1.5 for hydrostatic and leak test conditions during which the reactor core is not critical

At any time during the heatup or cooldown transient, K_{Ic} is determined by the metal temperature at the tip of a postulated flaw at the 1/4T and 3/4T location, the appropriate value for RT_{NDT}, and the reference fracture toughness curve. The thermal stresses resulting from the temperature gradients through the vessel wall are calculated and then the corresponding (thermal) stress intensity factors, K_{It}, for the reference flaw are computed. From Equation 2, the pressure stress intensity factors are obtained and, from these, the allowable pressures are calculated.

3/4.4.9 PRESSURE/TEMPERATURE LIMITS (Continued)

HEATUP (Continued)

During heatup, especially at the end of the transient, conditions may exist such that the effects of compressive thermal stresses and lower K_{Ic} values for steady-state and finite heatup rates do not offset each other and the pressure-temperature curve based on steady-state conditions no longer represents a lower bound of all similar curves for finite heatup rates when the 1/4T flaw is considered. Therefore, both cases have to be analyzed in order to assure that at any coolant temperature the lower value of the allowable pressure calculated for steady-state and finite heatup rates is obtained.

The second portion of the heatup analysis concerns the calculation of pressure-temperature limitations for the case in which a 1/4T flaw located at the 1/4T location from the outside surface is assumed. Unlike the situation at the vessel inside surface, the thermal gradients established at the outside surface during heatup produce stresses which are tensile in nature and thus tend to reinforce any pressure stresses present. These thermal stresses are dependent on both the rate of heatup and the time (or coolant temperature) along the heatup ramp. Since the thermal stresses at the outside are tensile and increase with increasing heatup rate, a lower bound curve cannot be defined. Rather, each heatup rate of interest must be analyzed on an individual basis.

Following the generation of pressure-temperature curves for both the steady-state and finite heatup rate situations, the final limit curves are produced as follows. A composite curve is constructed based on a point-by-point comparison of the steady-state and finite heatup rate data. At any given temperature, the allowable pressure is taken to be the lesser of the three values taken from the curves under consideration. The use of the composite curve is necessary to set conservative heatup limitations because it is possible for conditions to exist wherein, over the course of the heatup ramp, the controlling condition switches from the inside to the outside and the pressure limit must at all times be based on analysis of the most critical criterion.

~~Finally, the composite curves for the heatup rate data and the cooldown rate data are adjusted for possible errors in the pressure and temperature sensing instruments by the values indicated on the respective curves.~~

3/4.4.9 PRESSURE/TEMPERATURE LIMITS (Continued)

HEATUP (Continued)

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 3.4-4

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. ~~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.~~ 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-15745~~ WCAP-17441-NP, Revision 0, "Seabrook Unit 1 Heatup and Cooldown Limit Curves for Normal Operation", dated ~~December 2001.~~ October 2011

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

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

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

9. Westinghouse WCAP-17444-NP, Revision 0. "Reactor Vessel Closure Head/Vessel Flange Requirements Evaluation for Seabrook Unit 1." October 2011

3/4.4.9 PRESSURE/TEMPERATURE LIMITS (Continued)

COLD OVERPRESSURE PROTECTION (Continued)

during the following design basis transients

The OPERABILITY of two PORVs, or two RHR suction relief valves, or a combination of a PORV and RHR suction relief valve, or an RCS vent opening of at least 1.58 square inches ensures that the RCS will be protected from pressure transients which could exceed the limits of Appendix G to 10 CFR Part 50 when one or more of the RCS cold legs are less than or equal to 290°F. Either PORV or either RHR suction relief valve has adequate relieving capability to protect the RCS from overpressurization ~~when the transient is limited to either~~: (1) the start of an idle RCP with the secondary water temperature of the steam generator less than or equal to 50°F above the RCS cold leg temperatures, or (2) the start of a centrifugal charging pump and its injection into a water-solid RCS.

The Maximum Allowed PORV Setpoint for the Cold Overpressure Mitigation System (COMS) is derived by analysis which models the performance of the COMS assuming various mass input and heat input transients. Operation with a PORV Setpoint less than or equal to the maximum Setpoint ensures that Appendix G criteria will not be violated with consideration for: (1) a maximum pressure overshoot beyond the PORV Setpoint which can occur as a result of time delays in signal processing and valve opening; (2) a 50°F heat transport effect made possible by the geometrical relationship of the RHR suction line and the RCS wide range temperature indicator used for COMS; (3) instrument uncertainties; and (4) single failure. To ensure mass and heat input transients more severe than those assumed cannot occur, Technical Specifications require both Safety injection pumps and all but one centrifugal charging pump to be made inoperable while in MODES 4, 5, and 6 with the reactor vessel head installed and not fully detensioned, and disallow start of an RCP if secondary coolant temperature is more than 50°F above reactor coolant temperature. Exceptions to these requirements are acceptable as described below.

Operation above 350°F but less than 375°F with only one centrifugal charging pump OPERABLE and no Safety Injection pumps OPERABLE is allowed for up to 4 hours. As shown by analysis, LOCAs occurring at low temperature, low pressure conditions can be successfully mitigated by the operation of a single centrifugal charging pump and a single RHR pump with no credit for accumulator injection. Given the short time duration and the condition of having only one centrifugal charging pump OPERABLE and the probability of a LOCA occurring during this time, the failure of the single centrifugal charging pump is not assumed.

Operation below 350°F but greater than 325°F with all centrifugal charging and Safety Injection pumps OPERABLE is allowed for up to 4 hours. During low pressure, low temperature operation all automatic Safety Injection actuation signals except Containment Pressure-High are blocked. In normal conditions, a single failure of the