North Atlantic

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The Northeast Utilities System

October 11, 2002

Docket No. 50-443
NYN-02093

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

Seabrook Station
License Amendment Request 02-04
"Revision To Technical Specifications Associated With Pressure/Temperature Curves and Low Temperature Overpressure Protection Limits"

North Atlantic Energy Service Corporation (North Atlantic) has enclosed herein License Amendment Request (LAR) 02-04. License Amendment Request 02-04 is submitted pursuant to the requirements of 10 CFR 50.90 and 10 CFR 50.4.

Provided in Enclosure 1, LAR 02-04 proposes changes to the Seabrook Station Technical Specifications 3.4.9.1, "Reactor Coolant System – Pressure/Temperature Limits" and 3.4.9.3, "Reactor Coolant System – Overpressure Protection Systems". Specifically, the proposed changes will replace Technical Specification Figure 3.4-2, "Reactor Coolant System Heatup Limitations", Figure 3.4-3, "Reactor Coolant System Cooldown Limitations" and Figure 3.4-4, "RCS Cold Overpressure Protection Setpoints" to allow operation to 20 Effective Full Power Years.

A similar submittal was approved for Arkansas Nuclear One, Unit 2, on April 15, 2002 (TAC NOS. MB3301 and MB3302) and in part for Millstone Nuclear Power Station, Unit 3, on August 27, 2001 (TAC NO. MB1785).

The Index and the associated Bases for these Technical Specifications will be modified as a result of the proposed changes.

In addition, pursuant to 10 CFR 50.12, North Atlantic requests an exemption from the specific requirements of 10 CFR 50.60(a) and 10 CFR 50 Appendix G, based on American Society of Mechanical Engineers Code Case N-641, to support the revised reactor vessel analyses. Enclosure 1, LAR Section I.B contains the justification for the exemption request. A similar exemption was approved for Arkansas Nuclear One, Unit 2, on April 2, 2002 (TAC NO. MB3301).

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The Station Operation Review Committee and the Nuclear Safety Audit Review Committee have reviewed LAR 02-04.

As discussed in Enclosure 1, LAR Section IV, the proposed change does not involve a significant hazard consideration pursuant to 10 CFR 50.92. A copy of this letter and the enclosed LAR has been forwarded to the New Hampshire State Liaison Officer pursuant to 10 CFR 50.91(b). North Atlantic requests NRC Staff review of LAR 02-04, and issuance of a license amendment by September 30, 2003 (see Enclosure 1, Section V enclosed).

North Atlantic has determined that LAR 02-04 meets the criterion of 10 CFR 51.22(c)(9) for a categorical exclusion from the requirements for an Environmental Impact Statement (see Enclosure 1, Section VI enclosed).

Should you have any questions regarding this letter, please contact Mr. James M. Peschel, Manager - Regulatory Programs, at (603) 773-7194.

Very truly yours,
NORTH ATLANTIC ENERGY SERVICE CORP.

Joe M. Vargas
Director - Engineering

cc: H. J. Miller, NRC Region I Administrator
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ENCLOSURE 1 TO NYN-02093

North Atlantic

SEABROOK STATION UNIT 1

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

License Amendment Request 02-04,
"Revision To Technical Specifications Associated With Pressure/Temperature Curves and Low Temperature Overpressure Protection Limits"

This License Amendment Request is submitted by North Atlantic Energy Service Corporation pursuant to 10CFR50.90. The following information is enclosed in support of this License Amendment Request:

- **Section I - Introduction and Safety Assessment for Proposed Changes**
- **Section II - Markup of Proposed Changes**
- **Section III - Retype of Proposed Changes**
- **Section IV - Determination of Significant Hazards for Proposed Changes**
- **Section V - Proposed Schedule for License Amendment Issuance And Effectiveness**
- **Section VI - Environmental Impact Assessment**

I, Joe M. Vargas, Director – Engineering of North Atlantic Energy Service Corporation hereby affirm that the information and statements contained within this License Amendment Request are based on facts and circumstances which are true and accurate to the best of my knowledge and belief.

MARILYN R. SULLIVAN
MY COMMISSION EXPIRES APR. 17, 2007
NOTARY PUBLIC
NEW HAMPSHIRE

Joe M. Vargas
Director - Engineering

Sworn and Subscribed before me this 11th day of October, 2002

Marilyn R. Sullivan
Notary Public

SECTION I

INTRODUCTION AND SAFETY ASSESSMENT FOR PROPOSED CHANGES

I. INTRODUCTION AND SAFETY ASSESSMENT OF PROPOSED CHANGES

A. Introduction and Description of Change

Reviewer's Note

Throughout this amendment request, three terms are used which are defined as follows to ease the reviewer's interpretation of the material presented.

LTOP – Low Temperature Overpressure Protection – This term is used when referring to the cold overpressure requirements of ASME Section XI Appendix G and 10CFR50 Appendix G.

COPPS – Cold Overpressure Protection System – This is used when referring to the cold overpressure setpoints in Seabrook Station Unit 1 Technical Specification 3.4.9.3. The title of the Technical Specification is "Overpressure Protection Systems."

COMS – Cold Overpressure Mitigating System – This is used when referring to the cold overpressure study performed by Framatome, which establishes the setpoints for COPPS.

LTOP is also the system name for the Seabrook system which implements the COPPS setpoints that come from the COMS study.

Use of ASME Code Case N-641

The proposed Technical Specification changes to modify the pressure-temperature (P/T) limits and overpressure protection system setpoints rely in part on the use of the American Society of Mechanical Engineers (ASME) Code Case N-641. The revised P/T limits, as specified in ASME Code Case N-641, use a higher allowable stress intensity factor, K_{IC} instead of K_{IR}, which results in higher allowable pressures. K_{IR} is a reference stress intensity factor, based on the lower bound values of K_{IC} and K_{IA}. P/T curves and overpressure protection system setpoints based on the K_{IC} curve will enhance overall plant safety by opening the P/T operating window, with the greatest safety benefit in the region of low temperature operations. In addition, enhanced safety during critical plant operational periods, heatup and cooldown evolutions is expected.

The primary safety benefits in opening the low temperature operating window are a reduction in the challenges to pressurizer power-operated relief valves (PORVs), and additional margin to maintain reactor coolant pump (RCP) net positive suction head (NPSH) requirements. In addition, the pressure undershoot due to the relief capacity of one PORV and the time delay for the valve to close after opening for pressure relief due to a Cold Over Pressure Protection System (COPPS) event can result in damage to the RCP seals due to inadequate seal differential pressure. Damage to the RCP seals can require an unplanned shutdown to replace the seals. By raising the COPPS setpoints at low reactor coolant system (RCS) temperatures, the likelihood of challenging the pressurizer PORVs will be reduced, and operation at higher pressures to provide additional margin for RCP seal protection will be allowed.

The P/T limits determined using ASME Code Case N-641 are less restrictive than the requirements of 10 CFR 50 Appendix G, Section IV.A.2.b, which requires the use of methods equivalent to those provided by Appendix G to ASME Section XI. Since ASME Section XI Code Case N-641 was employed in the development of the reactor vessel beltline P/T limits, an exemption to 10 CFR 50.60(a), based on ASME Code Case N-641, is required to support the proposed Technical Specification changes (Section III).

Pressure/Temperature Curve Revision

License Amendment Request (LAR) 02-04 proposes changes to the Seabrook Station Technical Specifications 3.4.9.1, "Reactor Coolant System – Pressure/Temperature Limits" and 3.4.9.3, "Reactor Coolant System – Overpressure Protection Systems". Specifically, the proposed changes will replace Technical Specification Figure 3.4-2, "Reactor Coolant System Heatup Limitations", Figure 3.4-3, "Reactor Coolant System Cooldown Limitations" and Figure 3.4-4, "RCS Cold Overpressure Protection Setpoints". The proposed change of the reactor vessel P/T limit curves is required because the existing curves referenced above are valid through the attainment of 11.1 Effective Full Power Years (EFPY). Based on current projections, 11.1 EFPY will be achieved early in the next operating cycle, Cycle No. 10, at the end of the fourth quarter of 2003. The revised reactor vessel P/T limit curves will remain valid for 20 EFPY as demonstrated in the analysis documented in Reference (1) and included as Enclosure 2. The associated Bases will be modified as necessary as result of the proposed changes. In addition to the proposed Technical Specification, North Atlantic requests an exemption to 10 CFR50.60(b), based on ASME Code Case N-641, Reference (2), to support the development of the revised reactor vessel P/T limit curves.

Overpressure Protection System Revision

License Amendment Request (LAR) 02-04 proposes a change to Seabrook Station Technical Specification 3.4.9.3 "Reactor Coolant System – Overpressure Protection Systems". Specifically, the proposed change replaces Technical Specification Figure 3.4-4, "RCS Cold Overpressure Protection Setpoints" and revises the COPPS arming temperature. The maximum PORV COPPS setpoints specified in Figure 3.4-4 and COPPS arming temperature are revised to reflect the higher allowable low temperature overpressure protection (LTOP) pressure limit afforded by the use of ASME Code Case N-641. The revised pressurizer PORV COPPS setpoints have been derived for operation of Seabrook Station's reactor vessel to a cumulative exposure of 20 EFPY, consistent with the proposed revisions to the P/T limits in Figures 3.4-2 and 3.4-3. Credit is taken for the fact that experience shows that LTOP events are most likely to occur at isothermal conditions. The LTOP allowable pressure limit for the COPPS is therefore taken to be the steady-state (isothermal) P/T pressure limit curve. The allowable pressure is limited below 150°F, to comply with the closure head/vessel flange region limitation on system pressure imposed by 10 CFR Part 50, Appendix G. The revised COPPS setpoints provide overpressure protection for the Seabrook reactor vessel and closure head/flange region in accordance with ASME Code Case N-641 and 10 CFR Part 50 Appendix G.

B. Evaluation of Proposed Changes

Use of ASME Code Case N-641

The following information provides the basis for the exemption request to 10CFR50.60 for use of American Society of Mechanical Engineers (ASME) Section XI Code Case N-641, "Alternate Pressure/Temperature Relationship and Low Temperature Overpressure Protection System Requirements, Section XI, Division I", in lieu of methods specified in 10CFR50, Appendix G.

The requested exemption will allow use of ASME Code Case N-641 to (a) determine stress intensity factors for postulated circumferential defects in circumferential welds, and for postulated axial defects in plates, forgings and axial welds, and (b) use the K_{IC} fracture toughness curve shown on ASME XI, Appendix A, Figure A-2200-1, in lieu of the K_{IA} fracture toughness curve of ASME XI, Appendix G, Figure G-2210-1, as the lower bound fracture toughness.

10CFR50.12 states that the Commission may grant an exemption from requirements contained in 10CFR50 provided that:

1. **<u>The requested exemption is authorized by law:</u>**

   No law exists which precludes the activities covered by this exemption request. 10CFR50.60(b) allows the use of alternatives to 10CFR50, Appendices G and H when an exemption is granted by the Commission under 10CFR50.12.

2. **<u>The requested exemption does not present an undue risk to the public health and safety:</u>**

   10CFR50, Appendix G, requires, in part, that Article G-2120 of ASME Section XI, Appendix G, be used to determine the maximum postulated defects in reactor pressure vessels when determining pressure/temperature (P/T) limits for the vessel. These limits are determined for normal operation and pressure test conditions.

   Article G-2120 specifies, in part, that the postulated defect be in the surface of the vessel material and normal to the direction of maximum stress. ASME Section XI, Appendix G, also provides a methodology to determine the stress intensity factors for a maximum postulated defect normal to the maximum stress. The purpose of this article is, in part, to prevent non-ductile fractures by providing procedures to identify the most limiting postulated fractures to be considered in the development of P/T limits.

   Due to progress made in non-destructive examination (NDE) techniques over the last thirty years, it is unlikely that undetected defects will be present in the beltline region of reactor vessels. It is further unlikely to have axial cracks originating from a circumferential weld perpendicular to the weld seam orientation in reactor vessels. Both experience and engineering studies indicate that the primary degradation mechanism affecting the beltline region of the reactor vessel is neutron embrittlement. No other service reduced degradation mechanism exists at a pressurized water reactor to cause a prior existing defect located in the beltline region of the reactor vessel to grow while in service. Based on these considerations, and the fact that the P/T limit for reactor operation is the limiting pressure for any of the materials in the vessel, it is not necessary to include additional conservatism in the assumed flaw orientation for circumferential welds. ASME Section XI, Code Case N-641, and a previous Section XI, Appendix Code change correct this inconsistency in assumed flaw orientation for circumferential welds in vessels when calculating operating P/T limits.

Code Case N-641 provides benefits in terms of calculating P/T limits by revising the Section XI, Appendix G reference flaw orientation for circumferential welds in reactor vessels. The reference flaw is a postulated flaw that accounts for the possibility of an existing defect that may have gone undetected during the fabrication process. When considering a reference flaw with respect to a weld, the reference flaw would represent any prior existing defect that may have been introduced during fabrication. Thus, the intended application of a reference flaw is to account for prior existing defects that could physically exist within the geometry of the weldment. The currently endorsed ASME Section XI, Appendix G approach mandates consideration of an axial reference flaw in circumferential welds for purposes of calculating P/T limits. Postulating the Appendix G reference flaw in a circumferential weld is physically unrealistic and overly conservative, because the length of the flaw is 1.5 times the vessel thickness, which is much longer than the width of the reactor vessel girth weld. The possibility that an axial flaw may extend from a circumferential weld into a plate/forging or axial weld is already adequately covered by the requirement that axial defects be postulated in plates/forging and axial welds.

ASME Code Case N-641 reflects fabrication and NDE experience by allowing consideration of maximum postulated defects oriented circumferentially within the welds. Code Case N-641 also provides appropriate procedures to determine limiting circumferential weld defects and associated stress intensity factors for use in developing P/T limits per ASME Section XI, Appendix G procedures. The procedures allowed by Code Case N-641 are conservative and provide a margin of safety in the development of P/T operating and pressure test limits that will prevent nonductile fractures.

The revised P/T limits and overpressure protection system limits being proposed for Seabrook Station have been developed using the K_{IC} fracture toughness curve shown on ASME Section XI, Appendix A, Figure A-2200-1, in lieu of the K_{IA} fracture toughness curve of ASME XI, Appendix G, Figure G-2210-1, as the lower bound for fracture toughness. Use of the K_{IC} curve in determining the lower bound fracture toughness in the development of P/T operating limits curve is more technically correct than the K_{IA} curve. The K_{IC} curve models the slow heatup and cooldown process of a reactor vessel.

Use of this approach is justified by the initial conservatism of the K_{IA} curve when the curve was codified in 1974. This initial conservatism was necessary due to limited knowledge of reactor pressure vessel materials over time and usage. Since 1974, additional knowledge has been gained about the effect of usage on reactor pressure vessel materials. The additional knowledge demonstrates the lower bound on fracture toughness provided by the KI curve provides a margin of safety that is adequate to protect the public health and safety from potential reactor pressure vessel failure.

The COMS analyses for Seabrook Station were also performed using the method provided in Code Case N-641. Use of Code Case N-641 methodology in the determination of the LTOP conditions is more technically correct than the generic value included in earlier versions of ASME Section XI and eliminates inconsistencies in the margin of safety between reactor vessels of various geometries. Code Case N-641 provides bounding reactor vessel low temperature integrity protection during LTOP design basis transients. The LTOP lift setpoint utilizes 100% of the pressure determined to satisfy Appendix G, Paragraph G-2215 of ASME Section XI, Division 1, as a design limit. The approach is justified by consideration of the overpressurization design basis events and the resulting margin to reactor vessel failure.

P/T limit curves based on Code Case N-641 will enhance overall plant safety by opening the pressure/temperature operating window with the greatest safety benefit in the region of low temperature operations. The primary safety benefit in opening the low temperature operating window is a reduction in the challenges to LTOP valves. The proposed P/T limits include restrictions on allowable operating conditions and equipment operability requirements to ensure that operating conditions are consistent with the assumptions of the accident analysis. Specifically, the reactor coolant system (RCS) pressure and temperature must be maintained within the heatup and cooldown rate dependent pressure/temperature limits that are specified in TS 3.4.9. Therefore, this exemption does not present an undue risk to the public health and safety.

3. **The requested exemption will not endanger the common defense and security:**

The common defense and security are not endangered by this exemption request.

4. **Special circumstances are present which necessitate the request for an exemption to the regulations of 10CFR50.60:**

Pursuant to 10CFR50.12(a)(2), the NRC will consider granting an exemption to the regulations if special circumstances are present. This exemption meets the special circumstances of Paragraphs:

10CFR50.12(a)(2)(ii) demonstrates that the underlying purpose of the regulation will continue to be achieved; (a)(2)(iii) would result in undue hardship or other cost that are significant if the regulation is enforced and; (a)(2)(v) will provide only temporary relief from the applicable regulation and the licensee has made good faith efforts to comply with the regulations.

10CFR50.12(a)(2)(ii), Underlying Purpose of the Regulation Will Continue to be Achieved:

The underlying purpose of 10CFR50, Appendix G and ASME Section XI, Appendix G, is to satisfy the requirement that the reactor coolant pressure boundary be operated in a manner having sufficient margin to ensure that, when stressed, the vessel boundary behaves in a non-brittle manner and the probability of a rapidly propagating failure is minimized. Accordingly, that the P/T operating and test curves provide adequate margin in consideration of the uncertainties in determining the effects of irradiation on material properties.

Application of Code Case N-641 to determine P/T operating and hydrostatic test limit curves per ASME Section XI, Appendix G, provides appropriate procedures to determine limiting maximum postulated defects and considering those defects in the P/T limits. This application of the code case maintains the margin of safety originally considered for reactor pressure vessel materials. Therefore, use of Code Case N-641, as described above, satisfies the underlying purpose of the ASME Code and the NRC regulations to ensure an acceptable level of safety.

<u>10CFR50.12(a)(2)(iii), Result In Undue Hardship or Other Cost:</u>

The P/T operating window is defined by the P/T operating and test curves developed in accordance with the ASME Section XI, Appendix G procedure. Continued operation with these more restrictive P/T curves without the relief provided by ASME Code Case N-641 would unnecessarily restrict the pressure/temperature and LTOP operating window for Seabrook Station. Use of Case N-641 will minimize the potential for reactor coolant pump (RCP) impeller cavitation wear while operating in the LTOP region and reduce the potential for inadvertent actuation of the LTOP relief valves. Use of ASME Code Case N-641 in the development of the proposed P/T curves and Overpressure Protection System setpoint and enable temperature alleviates any unwarranted burden. Implementation of the proposed P/T curves and LTOP parameters as allowed by ASME Code Case N-641 does not reduce the margin of safety originally considered by either the NRC or ASME.

Compliance with the specified requirements of 10CFR50.60 would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety. ASME Code Case N-641 allows:

- Postulation of a circumferential defect in circumferential welds is appropriate in lieu of requiring the defect to be oriented across the weld from one plate or forging to the adjoining plate or forging. This circumstance was not considered at the time ASME Section XI, Appendix G was developed and imposes restrictions on P/T operating limits beyond those originally contemplated.

- A reduction in the fracture toughness lower bound is appropriate in lieu of the ASME Section XI, Appendix G, in the determination of reactor coolant pressure/temperature limits. This proposed alternative is acceptable because the Code Case maintains the relative margin of safety commensurate with that which existed at the time ASME Section XI, Appendix G was approved in 1974. Therefore, application of Code Case N-641 for Seabrook Station will ensure an acceptable margin of safety. The approach is justified by consideration of the overpressurization design basis events and the resulting margin to reactor vessel failure.

10CFR50.12(a)(2)(v), Licensee Has Made Good Faith Efforts to Comply with the Regulations.

The exemption provides only temporary relief from the applicable regulation and North Atlantic has made a good faith effort to comply with the regulation. We request that the exemption be granted until such time that the NRC generically approves ASME Code Case N-641 for use by the nuclear industry.

Restrictions on allowable operating conditions and equipment operability requirements have been established to ensure that operating conditions are consistent with the assumptions of the accident analysis. Specifically, RCS pressure and temperature must be maintained within the heatup and cooldown rate dependent pressure/temperature limits that are specified in the proposed amendment. Therefore, this exemption does not present an undue risk to the public health and safety.

To summarize, these proposed alternatives are acceptable because the Code Case maintains the relative margin of safety commensurate with that which existed at the time ASME Section XI, Appendix G, was approved in 1974. Therefore, application of Code Case N-641 Seabrook Station will ensure an acceptable margin of safety.

Pressure/Temperature Curve Revision

The recalculated reactor vessel P/T limit curves for normal operation are valid through 20 EFPY. The technical justification and methodologies employed in their development are documented in WCAP-15745, Reference (1) and included in Enclosure 2. The P/T limit curves were generated using the most recent reactor vessel surveillance capsule data which is documented in Duke Engineering and Services Report DES-NFQA-98-01, Reference (3). North Atlantic has previously submitted DES-NFQA-98-01 via North Atlantic Letter NYN98078, Reference (4). Capsule Y, the second Seabrook Station Unit 1 Surveillance Capsule, was removed from the reactor vessel after completion of Operating Cycle No. 5 in May 1997. At that point in time, the reactor vessel had accrued 5.572 EFPY. Analyses of the neutron dosimetry determined that the capsule had received an average neutron fast fluence (E>1 Mev) of 1.15 X 10^{19} n/cm^2. At that rate, this is equivalent to the fluence that will be received at the reactor vessel inner diameter after approximately 15 EFPY of operation. The second reactor vessel material capsule specimens were destructively tested and evaluated using the methodologies prescribed in Regulatory Guide 1.99, Revision 2, Reference (5), for predicting beltline material radiation embrittlement. It was concluded that the plate and weld material upper shelf energies will be maintained above 50 ft-lb throughout reactor vessel life as required by 10 CFR50, Appendix G. Also, based on the surveillance capsule data, the adjusted RT_{NDT} values for the plate and weld material were within the two standard deviations of Regulatory Guide 1.99, Revision 2 predictions. As all the requisite criteria of Regulatory Guide 1.99, Revision 2 were satisfied, Reference (3) concluded that the surveillance data was credible and the beltline material was responding as empirically predicted.

The P/T limit curves derived in WCAP-15745 used the adjusted RT_{NDT} value 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 was determined by using the unirradiated reactor vessel fracture toughness properties, estimating the radiation induced ΔRT_{NDT} and adding margin. The unirradiated RT_{NDT} is designated as the higher of either the drop weight nil-ductility transition temperature (NDTT) or the temperature at which the material exhibits at least 50 ft-lb of impact energy and 35 mil lateral expansion minus 60°F. The P/T limit curves developed in WCAP-15745 used the NRC approved methodology documented in Reference (6) with the exception of the following:

- The fluence values used were calculated values, not best estimate fluence values.
- The K_{IC} critical stress intensities were used in lieu of the K_{IA} critical stress intensities. This methodology was taken from approved ASME Code Case N-641, Reference (2).
- The 1996 Edition of Appendix G to Section XI of ASME Code, Reference (7), was used rather than the 1989 Edition.

Overpressure Protection System Revision

Two design basis overpressure events are considered in the COPPS setpoint development: a heatup transient and a mass addition event. The COPPS design basis heatup transient starts from an initial condition where the RCS flow rate is at zero, the steam generators are at a temperature 50°F hotter than the rest of the RCS, the pressurizer is water solid (a bubble has not yet been formed) and then a RCP is inadvertently started. The starting of the RCP initiates heat transfer from the secondary to the primary side which causes a pressurization of the primary side. Simulations of the heatup event were used to determine the peak RCS pressure setpoint overshoot defined as the peak pressure in the vessel adjacent to the beltline minus the nominal PORV setpoint. Calculations were performed over a range of assumed initial water temperatures below the expected COPPS arming temperature.

The second design basis event is a mass addition event. The event considered is mass addition from a single charging pump via both the normal charging flow path, with the charging flow and head control valves fully open, and the charging pump safety injection flow path in parallel, due to an inadvertent opening of one of the two SI flow path block valves. Letdown flow was assumed to be isolated. Current plant Technical Specifications limit the number of operable charging pumps to a single Centrifugal Charging Pump (CCP) in Modes 4 and 5 except for a brief transition period when entering or exiting Mode 3 and/or when swapping pumps. Therefore the analysis assumed flow from a single CCP. Maximum setpoint overshoot was determined as a function of initial wide-range pressure and PORV setpoint for a cold (100°F) reactor coolant condition. Assuming a cold RCS conservatively minimizes the compressibility of the RCS water volume, maximizing the rate of pressure increase and setpoint pressure overshoot.

The pressure overshoot values calculated for the design basis heat and mass addition events were used to determine the maximum allowable PORV COPPS setpoint required to prevent violation of the LTOP allowable pressure limit during the event. The maximum PORV COPPS setpoint is lower than the LTOP allowable pressure limit minus the overshoot predicted for the applicable temperature condition. The analysis included consideration of transient temperature effects on the temperature-dependent COPPS setpoint and vessel beltline water temperature. To assure the COPPS setpoint will not exceed the maximum value specified in Figure 3.4-4 temperature and pressure measurement and instrument uncertainties are applied in determining the nominal PORV COPPS setpoint function hardware settings and COPPS arming temperature. The setpoints of the two PORVs are also staggered to minimize the probability of opening more than one valve, since one valve provides adequate relief flow. The recalculated COPPS setpoints are valid through 20 EFPY and provide overpressure protection for the Seabrook reactor vessel and closure head/flange region in accordance with ASME Code Case N-641 and 10 CFR Part 50 Appendix G.

The ability of the Residual Heat Removal (RHR) system suction line safety valves to provide protection of the revised LTOP pressure limit independent of the COPPS as allowed by Technical Specification 3.4.9.3.a was also verified.

Cold Overpressure Mitigation System (COMS) Arming Temperature

The COMS system design has an arming bistable for each Power-Operated Relief Valve (PORV). When wide-range (WR) RCS temperature goes below the selected arming temperature, the bistable arms COMS by causing the associated PORV block valve to open (if it is closed) and enabling the PORV COMS setpoint.

ASME Code Case N-641 indicates the LTOP System Effective Temperature is the "*temperature at or above which the safety relief valves provide adequate protection against nonductile failure*". Per the code case, LTOP systems shall be effective below the higher of an inlet coolant temperature of 200°F or a coolant temperature corresponding to a reactor vessel 1/4T metal temperature of RT_{NDT} + 40°F for inside axial surface flaws and RT_{NDT} − 85°F for inside circumferential surface flaws for all vessel beltline materials. The adjusted 1/4T RT_{NDT} for the limiting beltline material at 20 EFPY is 109°F. Therefore, the code case requires the COMS system to be effective below an inlet coolant temperature of 200°F.

The selected LTOP allowable pressure limit at 200°F, however, is lower than the safety relief valve setpoint (2560 psig = 2485 psig ±3%, per TS 3.4.2.1). The pressure limit corresponding to the potentially 50°F lower transient temperature in the reactor vessel (RV) downcomer during a mass addition event is also less than the maximum safety valve setpoint. Therefore, the COMS arming temperature is selected to be greater than or equal to the temperature (273.6°F) at which the allowable COMS relief valve pressure setpoint is ≥ the pressurizer safety relief valve maximum opening setpoint (2560 psig) plus an allowance for the difference between indicated WR pressure and pressure at the safety valve (approximately 67 psi)[1], i.e. > 2627 psig).

This temperature is adjusted to a higher value to include allowances for applicable temperature measurement and arming bistable uncertainties (13.3°F). Therefore the revised arming temperature is 273.6 + 13.3 = 286.9°F which is rounded up to 290°F. The nominal PORV COMS setpoints at the arming temperature would be: PCV-456A = 2670.5 psig, and 2610.5 psig for PCV-456B.

[1] From the PRESS model input, the elevation difference between the top of the pressurizer to the bottom of the reactor vessel = 84.55 ft, corresponding to an elevation head = 84.55 ft x 62 lb/ft^3. ÷ 144 in^2/ft^2 = 36.4 psi. The flow loss with 4 RCPs running is approximately 46 psi at 70°F. Thus the total pressure difference between the bottom of the vessel where WR pressure is tapped and the top of the pressurizer could be as high as 46 + 36.4 = 82.4 psi. The indicated WR pressure is approximately 15.3 psi lower than the pressure at the tap, so the final pressure difference = 82.4 – 15.3 = 67.1 psi.

C. Safety Assessment Conclusion of Proposed Changes

North Atlantic concludes that based upon the above discussion (Section B, Evaluation of Proposed Changes), as well as the "Determination of Significant Hazards for Proposed Changes," presented in Section IV, that the proposed changes do not adversely affect or endanger the health or safety of the general public or involve a significant safety hazard.

D. References

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

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. Duke Engineering and Services Report DES-NFQA-98-01, Revision 0, "Analysis of Seabrook Station Unit 1 Reactor Vessel Surveillance Capsules U and Y", dated May 1998.

4. North Atlantic Letter NYN-98078, T. C. Feigenbaum to U. S. NRC, "Seabrook Station Reactor Vessel Surveillance Capsule Report", dated June 5, 1998

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

6. Westinghouse WCAP-14040-NP-A, Revision 2, "Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves", dated January 1996.

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

8. Framatome Letter NFSB 02-0061, "COMS Setpoints for 20 EFPY," August 30, 2002.

9. "Seabrook Station Cold Overpressure Mitigating System (COMS) Setpoint Development Methodology," August 2002.

SECTION II

MARKUP OF PROPOSED CHANGES

Refer to the attached markup of the proposed changes to the Technical Specifications. The attached markup reflects the currently issued revision of the Technical Specifications listed below. Pending Technical Specifications or Technical Specification changes issued subsequent to this submittal are not reflected in the enclosed markup.

The following Technical Specification changes are included in the attached markup:

Technical Specification	Title	Page
Index	Limiting Conditions for Operation and Surveillance Requirements	vi & x & xi
3.4.9.1	Reactor Coolant System Pressure/Temperature Limits	3/4 4-31 & 32
3.4.9.3	Reactor Coolant System Overpressure Protection Systems	3/4 4-34 & 36
3/4.4.9	Pressure/Temperature Limits	B 3/4 4-7 through 9 B 3/4 4-11 through 15

REACTOR COOLANT SYSTEM

3/4.4.9 PRESSURE/TEMPERATURE LIMITS

For Information Only

GENERAL

LIMITING CONDITION FOR OPERATION

3.4.9.1 The Reactor Coolant System (except the pressurizer) temperature and pressure shall be limited in accordance with the limit lines shown on Figures 3.4-2 and 3.4-3 during heatup, cooldown, criticality, and inservice leak and hydrostatic testing with:

a. A maximum heatup of 100°F in any 1-hour period,

b. A maximum cooldown of 100°F in any 1-hour period, and

c. A maximum temperature change of less than or equal to 10°F in any 1-hour period during inservice hydrostatic and leak testing operations above the heatup and cooldown limit curves.

APPLICABILITY: At all times.

ACTION:

With any of the above limits exceeded, restore the temperature and/or pressure to within the limit within 30 minutes; perform an engineering evaluation to determine the effects of the out-of-limit condition on the structural integrity of the Reactor Coolant System; determine that the Reactor Coolant System remains acceptable for continued operation or be in at least HOT STANDBY within the next 6 hours and reduce the RCS T_{avg} and pressure to less than 200°F and 500 psig, respectively, within the following 30 hours.

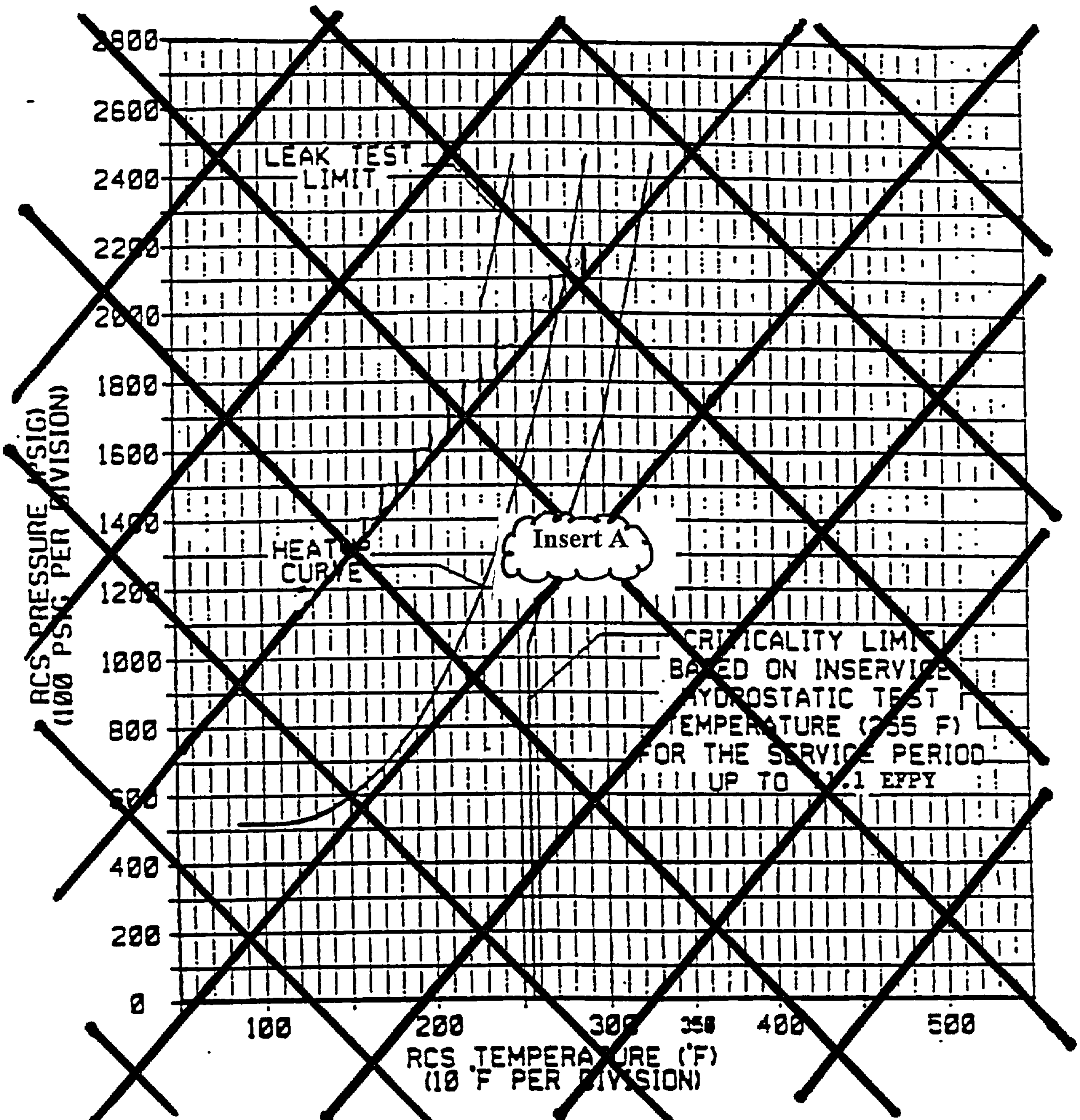
SURVEILLANCE REQUIREMENTS

4.4.9.1 The Reactor Coolant System temperature and pressure shall be determined to be within the limits at least once per 30 minutes during system heatup, cooldown, and inservice leak and hydrostatic testing operations.

7C4 91 190

~~Controlling material:~~	~~Base metal~~
~~Copper content:~~	~~0.06 WT%~~
~~RT_{NDT} initial:~~	~~40°F~~
~~RT_{NDT} after 11.1 EFPY:~~	~~1/4T, 100°F~~
	~~3/4T, 86°F~~

~~Curve applicable for heatup rates up to 60°F/hr for the service period up to 11.1 EFPY and contains margins of 10°F and 60 psig for possible instrument errors~~



~~FIGURE 3.4-2~~

~~REACTOR COOLANT SYSTEM HEATUP LIMITATIONS - APPLICABLE UP TO 11.1 EFPY~~

Insert A, Page 3/4 4-31

MATERIAL PROPERTY BASIS

Limiting material: LOWER SHELL PLATE R-1808-1
Limiting ART values at 20 EFPY:1/4T, 109°F
3/4T, 88°F

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

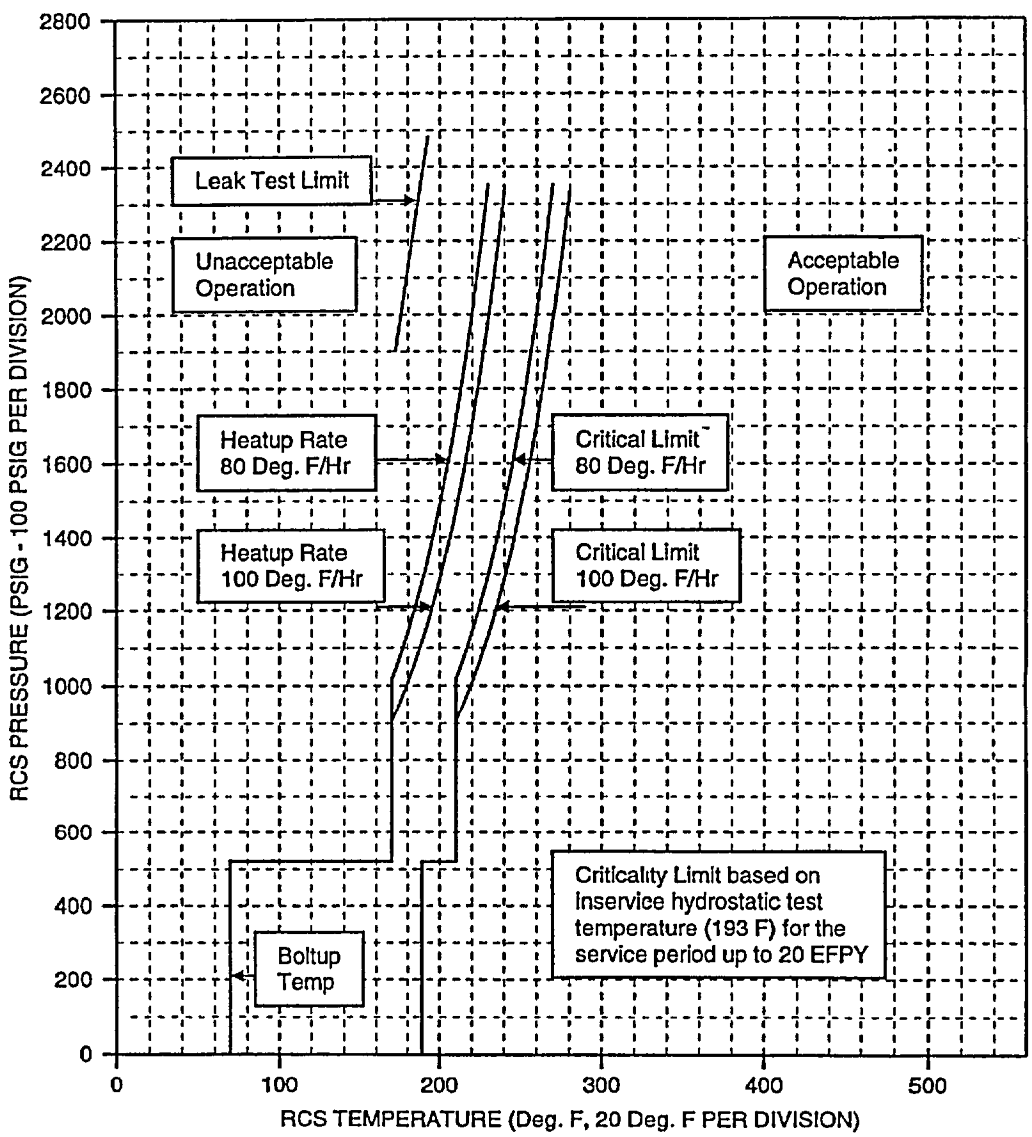


FIGURE 3.4-2
REACTOR COOLANT SYSTEM HEATUP LIMITATIONS - APPLICABLE UP TO 20 EFPY

~~MATERIAL PROPERTY BASIS~~

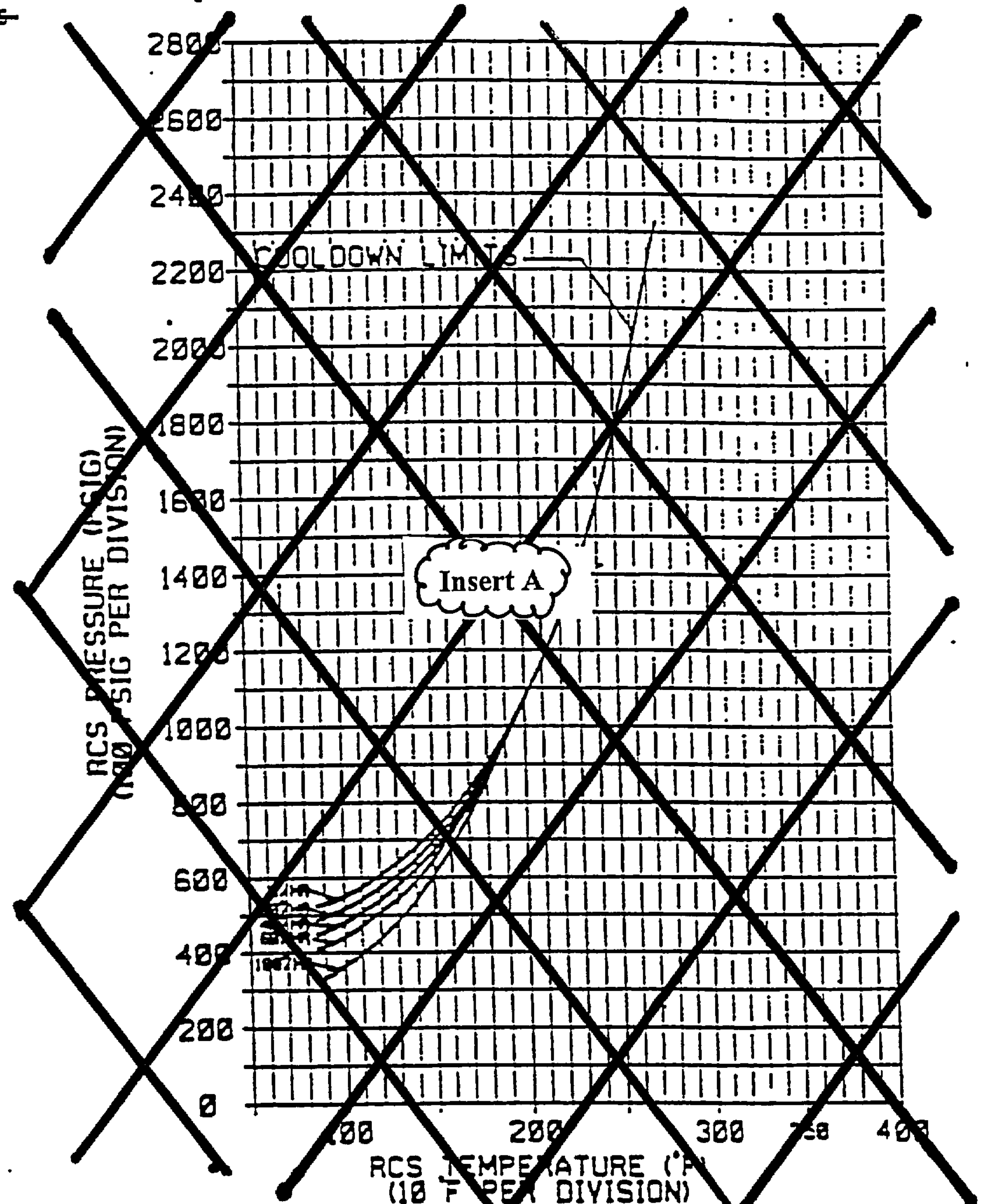
~~Controlling material:~~ ~~Base metal~~

~~Copper content:~~ ~~0.06 WT%~~

~~RT_{NDT} initial:~~ ~~40°F~~

~~RT_{NDT} after 11.1 EFPY:~~ ~~1/4T, 108°F~~
~~3/4T, 86°F~~

~~Curve applicable for cooldown rates up to 100°F/hr for the service period up to 11.1 EFPY and contains margins of 10°F and 60 psig for possible instrument errors~~



~~FIGURE 3.4-3~~

~~REACTOR COOLANT SYSTEM COOLDOWN LIMITATIONS - APPLICABLE UP TO 11.1 EFPY~~

Insert A, Page 3/4 4-32

MATERIAL PROPERTY BASIS

Limiting material: LOWER SHELL PLATE R-1808-1
Limiting ART values at 20 EFPY:1/4T, 109°F
3/4T, 88°F

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

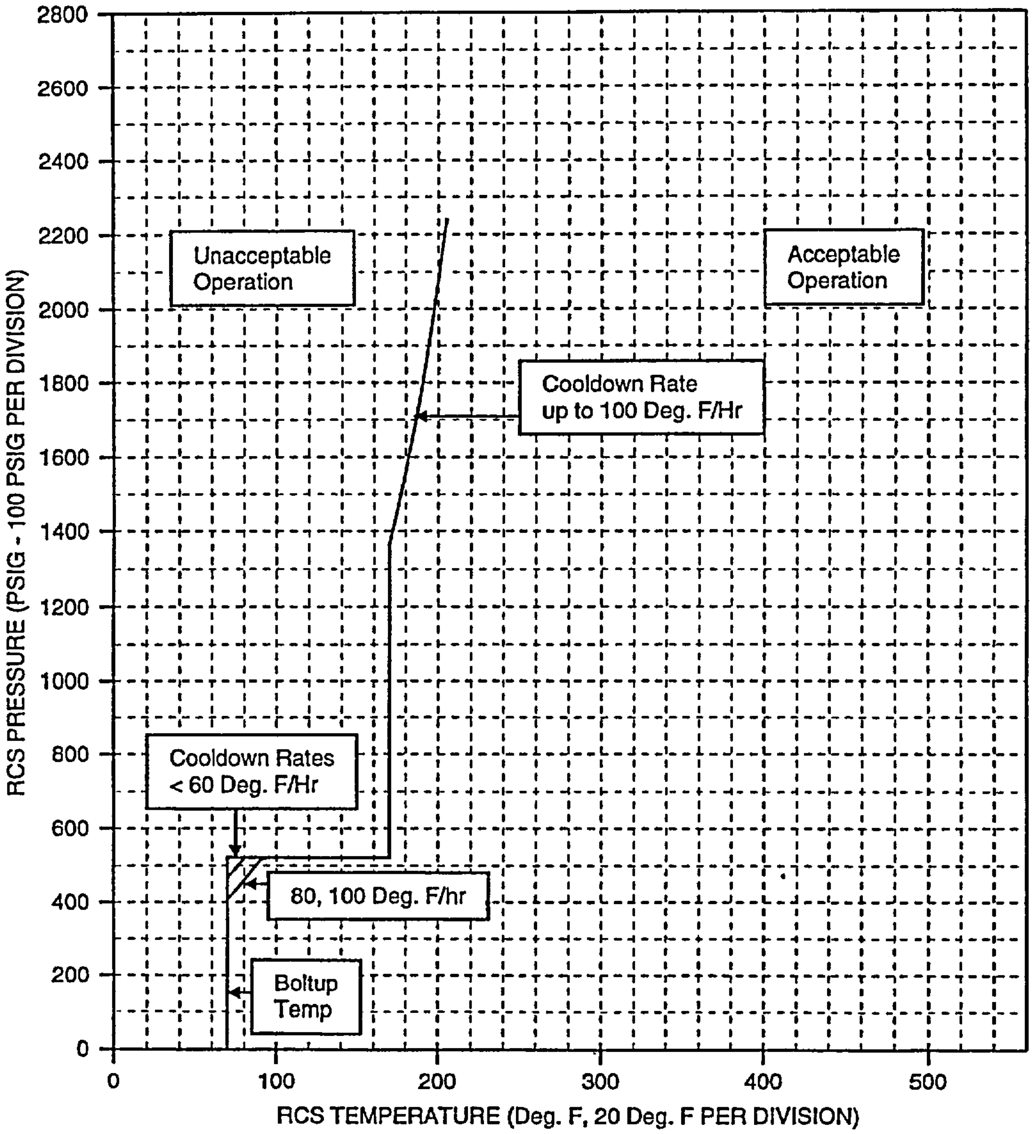


FIGURE 3.4-3
REACTOR COOLANT SYSTEM COOLDOWN LIMITATIONS - APPLICABLE UP TO 20 EFPY

LIMITING CONDITION FOR OPERATION

3.4.9.3 The following Overpressure Protection Systems shall be OPERABLE:

a. In MODE 4 when the temperature of any RCS cold leg is less than or equal to ~~329~~ 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*.

APPLICABILITY: MODE 4 when the temperature of any RCS cold leg is less than or equal to ~~329~~ 290°F; MODE 5 and MODE 6 with the reactor vessel head on and the vessel head closure bolts not fully detensioned.

ACTION:

a) In MODE 4 with all Safety Injection pumps inoperable and with one of the two required overpressure protection devices inoperable, either restore two overpressure protection devices to OPERABLE status within 7 days or within the next 8 hours

   (a) depressurize the RCS and

   (b) vent the RCS through at least a 1.58-square-inch vent.

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

For Information Only

3.4.9.3

ACTION: (Continued)

b) In MODE 5 and MODE 6 with all Safety Injection pumps inoperable and with one of the two required overpressure protection devices inoperable, restore two overpressure protection devices to OPERABLE status within 24 hours or within the next 8 hours

(a) depressurize the RCS and

(b) vent the RCS through at least a 1.58-square-inch vent.

c) In MODE 4, MODE 5 and MODE 6 with all Safety Injection pumps inoperable and with both of the two required overpressure protection devices inoperable, within the next 8 hours

(a) depressurize the RCS and

(b) vent the RCS through at least a 1.58-square-inch vent.

d) In the event the PORVs, or the RHR suction relief valves, or the RCS vent(s) are used to mitigate an RCS pressure transient, a Special Report shall be prepared and submitted to the Commission pursuant to Specification 6.8.2 within 30 days. The report shall describe the circumstances initiating the transient, the effect of the PORVs, or the RHR suction relief valves, or RCS vent(s) on the transient, and any corrective action necessary to prevent recurrence.

e) In MODE 5 and MODE 6 with all Safety Injection pumps except one inoperable and with the RCS vent area less than 18 square inches or RCS water level not in a reduced inventory condition, immediately restore all Safety Injection pumps to inoperable status.

For Information Only

OVERPRESSURE PROTECTION SYSTEMS

SURVEILLANCE REQUIREMENTS

4.4.9.3.1 Each PORV shall be demonstrated OPERABLE when the PORV(s) are being used for overpressure protection by:

a. Performance of an ANALOG CHANNEL OPERATIONAL TEST on the PORV actuation channel, but excluding valve operation, at least once per 31 days thereafter when the PORV is required OPERABLE; and

b. Performance of a CHANNEL CALIBRATION on the PORV actuation channel at least once per 18 months; and

c. Verifying the PORV isolation valve is open at least once per 72 hours.

4.4.9.3.2 Each RHR suction relief valve shall be demonstrated OPERABLE when the RHR suction relief valve(s) are being used for overpressure protection as follows:

a. For RHR suction relief valve RC-V89 by verifying at least once per 72 hours that RHR suction isolation valves RC-V87 and RC-V88 are open.

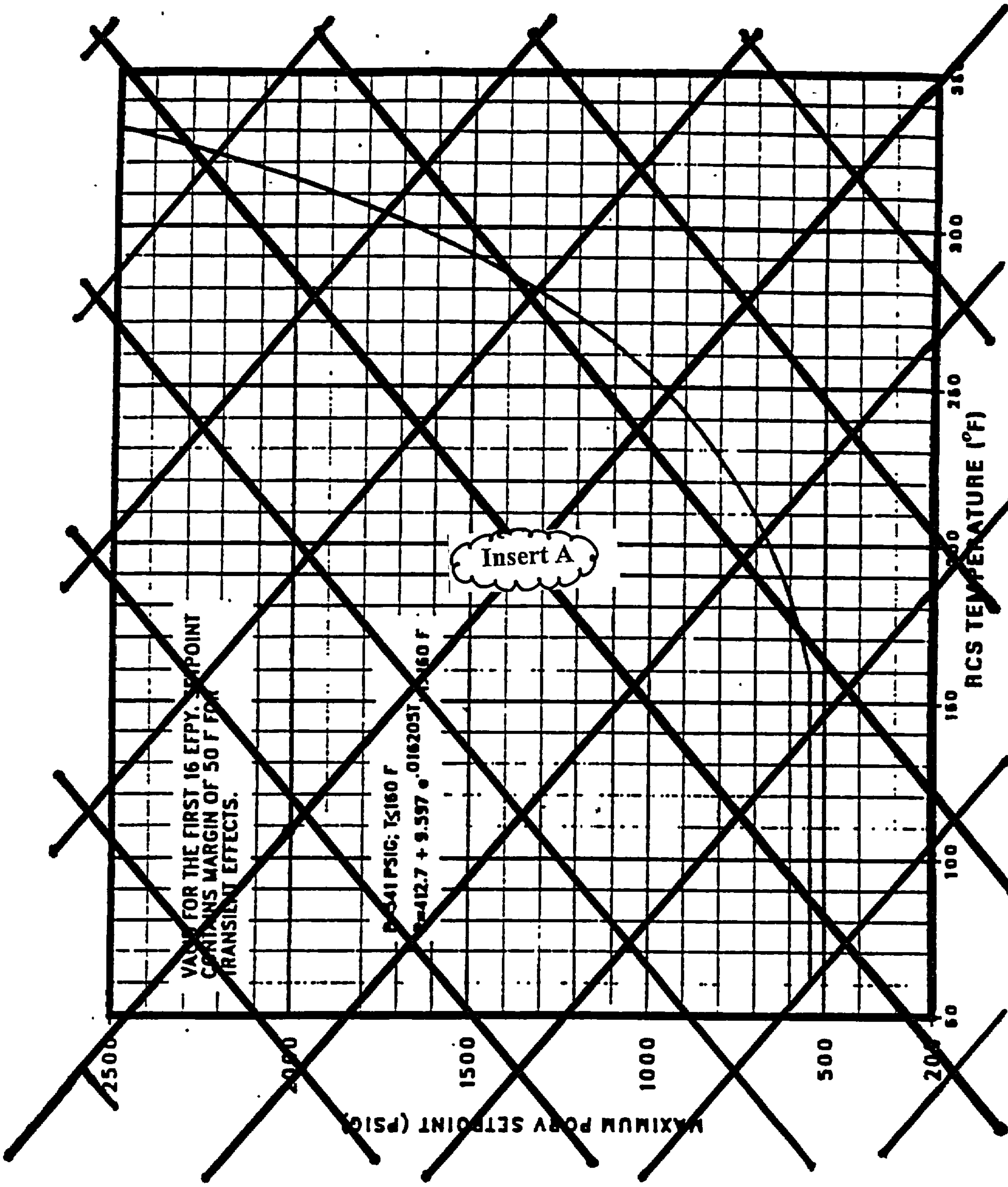
b. For RHR suction relief valve RC-V24 by verifying at least once per 72 hours that RHR suction isolation valves RC-V22 and RC-V23 are open.

c. Testing pursuant to Specification 4.0.5.

4.4.9.3.3 The RCS vent(s) shall be verified to be open at least once per 12 hours** when the vent(s) is being used for overpressure protection.

4.4.9.3.4 The reactor vessel water level shall be verified to be lower than 36 inches below the reactor vessel flange at least once per 12 hours when the reduced inventory condition is being used for overpressure protection.

**Except when the vent pathway is provided with a valve(s) or device(s) that is locked, sealed, or otherwise secured in the open position, then verify this valve(s) or device(s) open at least once per 31 days.



~~FIGURE 3.4-4 RCS COLD OVERPRESSURE PROTECTION SETPOINTS~~

Amendment No.

Insert A, Page 3/4 4-36

VALID FOR THE FIRST 20 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

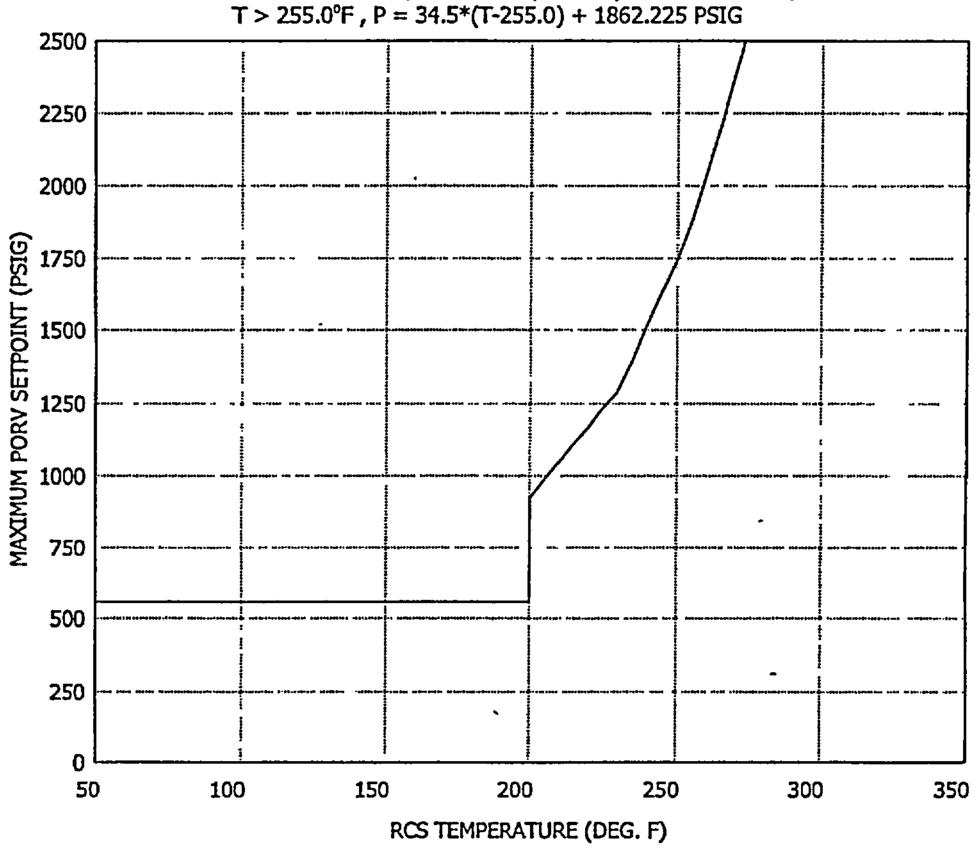


FIGURE 3.4-4 RCS COLD OVERPRESSURE PROTECTION SETPOINTS

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 ~~III~~ 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 ofnon-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.

Insert A

~~The fracture toughness properties of the ferritic materials in the reactor vessel are determined in accordance with the NRC Regulatory Guide 1.99, Revision 2, and in accordance with additional reactor vessel requirements. These properties are then evaluated in accordance with Appendix G of the 1972 Winter Addenda to Section III of the ASME Boiler and Pressure Vessel Code and the calculation methods described in WCAP-7924-A, "Basis for Heatup and Cooldown Limit Curves," April 1975~~.

Insert B

Heatup and cooldown limit curves are calculated using the ~~most limiting value of the nil-ductility reference temperature, RT~~$_{NDT}$~~, at the end of 11.1 effective full power years (EFPY) of service life. The 11.1 EFPY service life period~~

Insert A, Page B3/4 4-7

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.

The P/T limits have been established in accordance with the requirements of ASME Boiler and Pressure Vessel Code Section XI, Appendix G, as modified by ASME Code Case N-641, Reference (2), and the additional requirements of 10CFR50 Appendix G, Reference (3). The heatup and cooldown P/T limit curves for normal operation, Figures 3.4-2 and 3.4-3 respectively, are valid for a service period of 20 effective full power years (EFPY). The technical justification and methodologies utilized in their development are documented in WCAP-15745, Reference (4). The P/T curves were generated based on the latest available reactor vessel information and latest calculated fluences.

Insert B, Page B3/4 4-7

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

3/4.4.9 PRESSURE/TEMPERATURE LIMITS (Continued)

~~is chosen such that the limiting RT_{NDT} at the 1/4T location in the core region is greater than the RT_{NDT} of the limiting unirradiated material. The selection of such a limiting RT_{NDT} assures that all components in the Reactor Coolant System will be operated conservatively in accordance with applicable Code requirements.~~

best estimate

limiting beltline

surveillance capsule data

The reactor vessel materials have been tested to determine their initial RT_{NDT}~~; the results of these tests are shown in Table B 3/4.4-1.~~ 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, copper ~~content,~~ and nickel content of the material ~~in question,~~ can be predicted using ~~Figure B 3/4.4-1~~ and the value of ΔRT_{NDT} computed by Regulatory Guide 1.99, Revision 2~~, "Radiation Embrittlement of Reactor Vessel Materials."~~ The heatup and cooldown limit curves of Figures 3.4-2 and 3.4-3 include predicted adjustments for this shift in RT_{NDT} at the end of ~~11.1~~ 20 EFPY as well as adjustments for possible errors in the pressure and temperature sensing instruments. Insert A

U and Y were

~~Values of ΔRT_{NDT} determined in this manner may be used until~~ the results from the material surveillance program, evaluated according to ASTM E185, ~~are available~~. Capsules ~~will be~~ 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 ~~can be~~ used to predict future radiation damage to the reactor vessel material by using the lead factor and the withdrawal time of the capsule. ~~Evaluation of surveillance capsule data will be conducted in accordance with NRC Regulatory Guide 1.99, Revision 2.~~

~~Allowable pressure-temperature relationships for various heatup and cooldown rates are calculated using methods derived from Appendix G in Section III of the ASME Boiler and Pressure Vessel Code as required by Appendix G to 10 CFR Part 50, and these methods are discussed in detail in WCAP-7924-A.~~

~~The general method for calculating heatup and cooldown limit curves is based upon the principles of the linear elastic fracture mechanics (LEFM) technology. In the calculation procedures, a semielliptical surface defect with a depth of one-quarter of the wall thickness, T, and a length of 3/2T~~

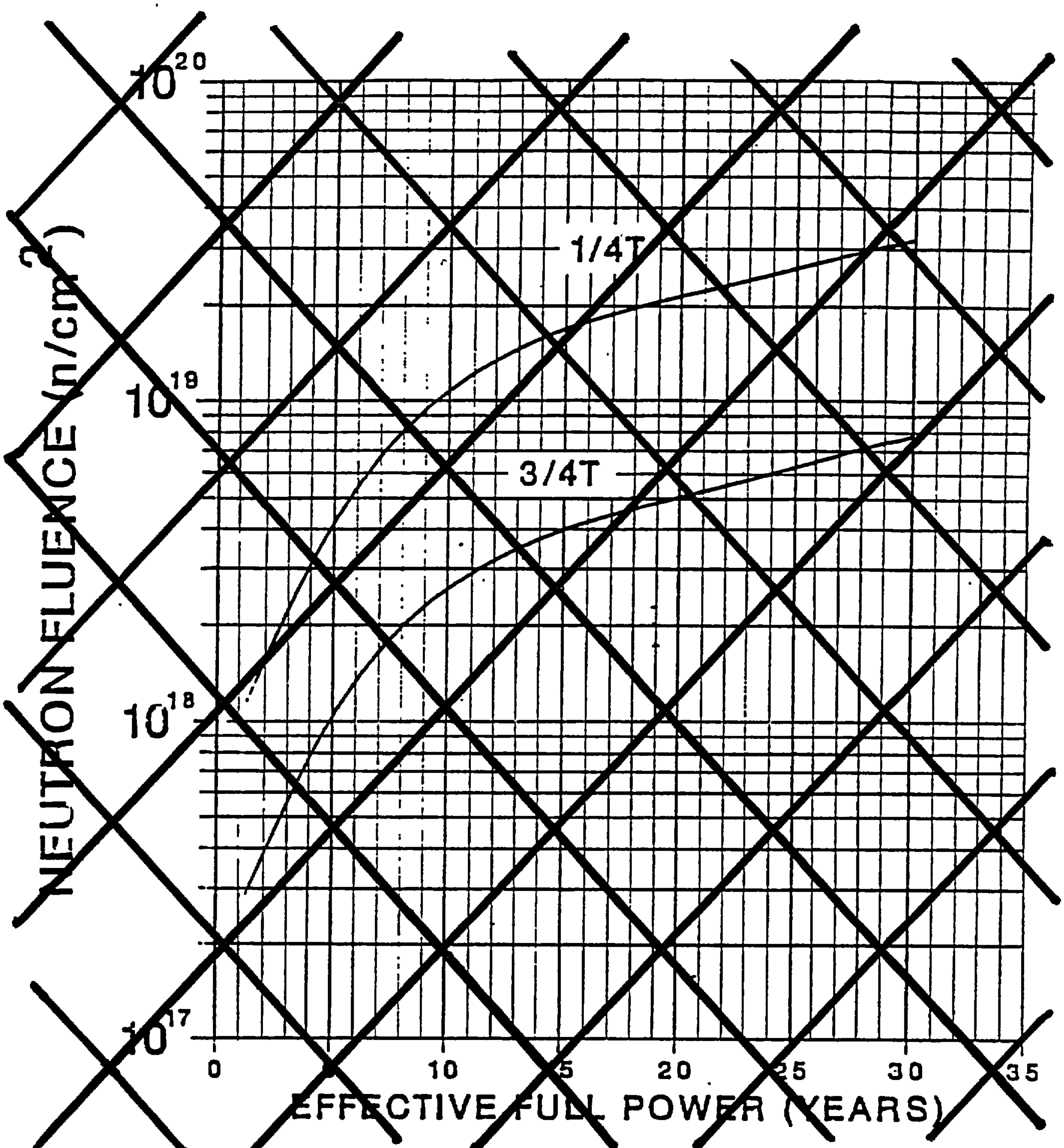
Insert B

Insert A, Page B3/4 4-8

Surveillance capsule data, documented in Reference (6), is available for two capsules (Capsules U and Y) having already been removed from the reactor vessel. This surveillance capsule data was used to calculate chemistry factor (CF) values per Position 2.1 of Regulatory Guide 1.99, Revision 2. It also noted that Reference (6) concluded that all the surveillance data was credible as the beltline material was behaving as empirically predicted.

Insert B, Page B3/4 4-8

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 (6). 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.



(THIS FIGURE NUMBER IS NOT USED)

Amendment No.

FIGURE B 3/4.4-1

~~FAST NEUTRON FLUENCE (E>1MeV) AS A FUNCTION OF FULL POWER SERVICE LIFE~~

~~13Cb~~ ~~91~~ ~~36~~

TABLE B 3/4.4-1 (THIS TABLE NUMBER IS NOT USED)

~~REACTOR VESSEL TOUGHNESS~~

Component	Code No.	Material Spec. No.	Cu (%)	P (%)	T_{NDT} (°F)	RT_{NDT} (°F)	Avg. Shelf Energy NMWD* (ft-lb)	Avg. Shelf Energy MWD** (ft-Lb)
Closure Head Dome	R1809-1	A533B,CL.1	0.15	0.011	-40	10	80.5	-
Closure Head Torus	R1810-1	A533B,CL.1	0.08	0.012	-50	0	104	-
Closure Head Flange	R1802-1	A508,CL.2	-	0.013	10	10	105.5	-
Vessel Flange	R1801-1	A508,CL.2	-	0.012	20	30	91	-
Inlet Nozzle	R1804-1	A508,CL.2	0.10	0.011	0	0	125	-
Inlet Nozzle	R1804-2	A508,CL.2	0.09	0.010	-20	-20	125	-
Inlet Nozzle	R1804-3	A508,CL.2	0.08	0.010	-20	-20	131	-
Inlet Nozzle	R1804-4	A508,CL.2	0.10	0.013	-20	-20	128	-
Outlet Nozzle	R1805-1	A508,CL.2	-	0.005	-20	-10	115	-
Outlet Nozzle	R1805-2	A508,CL.2	-	0.004	-20	-20	132	-
Outlet Nozzle	R1805-3	A508,CL.2	-	0.009	-10	-10	128	-
Outlet Nozzle	R1805-4	A508,CL.2	-	0.005	-10	-10	117	-
Nozzle Shell	R1807-1	A533B,CL.1	0.08	0.011	-30	30	66	-
Nozzle Shell	R1807-2	A533B,CL.1	0.09	0.012	-40	30	66.5	-
Nozzle Shell	R1807-3	A533B,CL.1	0.06	0.010	-20	10	107	-
Inter. Shell	R1806-1	A533B,CL.1	0.04	0.012	-30	40	82	139.5
Inter. Shell	R1806-2	A533B,CL.1	0.05	0.007	-30	0	102	143.5
Inter. Shell	R1806-3	A533B,CL.1	0.07	0.007	-40	10	115	138
Lower Shell	R1808-1	A533B,CL.1	0.05	0.005	-30	40	78	128.5
Lower Shell	R1808-2	A533B,CL.1	0.05	0.007	-20	10	77	127
Lower Shell	R1808-3	A533B,CL.1	0.06	0.007	-20	40	78	130.5
Bottom Head Torus	R1811-1	A533B,CL.1	0.15	0.010	-50	0	94.5	-
Bottom Head Dome	R1812-1	A533B,CL.1	0.09	0.009	-30	0	97.5	-
Inter. & Lower Shell Long Weld Seams	G1.72	Sub Arc Weld	0.07	0.008	-50	-50	200	-
Inter. & Lower Shell Girth Weld Seam	G1.72	Sub Arc Weld	0.07	0.008	-50	-50	200	-

*NMWD - Normal to Major Working Direction
**MWD - Major Working Direction

3/4.4.9 PRESSURE/TEMPERATURE LIMITS (Continued)

is assumed to exist at the inside of the vessel wall as well as at the outside of the vessel wall. The dimensions of this postulated crack, referred to in Appendix G of ASME Section III as the reference flaw, amply exceed the current capabilities of inservice inspection techniques. Therefore, the reactor operation limit curves developed for this reference crack are conservative and provide sufficient safety margins for protection against nonductile failure. To assure that the radiation embrittlement effects are accounted for in the calculation of the limit curves, the most limiting value of the nil-ductility reference temperature, RT_{NDT}, is used and this includes the radiation-induced shift, ΔRT_{NDT}, corresponding to the end of the period for which heatup and cooldown curves are generated.

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_{IR}, for the metal temperature at that time. K_{IR} is obtained from the reference fracture toughness curve, defined in Appendix G to the ASME Code. The K_{IR} curve is given by the equation:

$$K_{IR} = 26.78 + 1.223 \exp [0.0145(T-RT_{NDT} + 160)] \quad (1)$$

Where: K_{IR} is the reference stress intensity factor as a function of the metal temperature T and the metal nil-ductility reference temperature RT_{NDT}. Thus, the governing equation for the heatup-cooldown analysis is defined in Appendix G of the ASME Code as follows:

$$C\, K_{IM} + K_{It} \leq K_{IR} \quad (2)$$

Where: K_{IM} = the stress intensity factor caused by membrane (pressure) stress,

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

K_{IR} = constant provided by the Code as a function of temperature relative to the RT_{NDT} of the material,

C = 2.0 for level A and B service limits, and

C = 1.5 for inservice hydrostatic and leak test operations.

At any time during the heatup or cooldown transient, K_{IR} is determined by the metal temperature at the tip of the postulated flaw, the appropriate value

Insert A, Page B3/4 4-12

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:

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

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 Class1, 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:

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

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)

~~resulting from temperature gradients through the vessel wall are calculated and then the corresponding thermal stress intensity factor,~~ $\cancel{K_{It}}$, ~~for the reference flaw is computed. From Equation (2) the pressure stress intensity factors are obtained and, from these, the allowable pressures are calculated~~.

COOLDOWN

For the calculation of the allowable pressure versus coolant temperature during cooldown, the ~~Code~~ reference flaw of Appendix G to the ASME Code is assumed to exist at the inside of the vessel wall. During cooldown, the controlling location of the flaw is always at the inside of the wall because the thermal gradients produce tensile stresses at the inside, which increase with increasing cooldown rates. Allowable pressure-temperature relations are generated for both steady-state and finite cooldown rate situations. From these relations, composite limit curves are constructed for each cooldown rate of interest.

The use of the composite curve in the cooldown analysis is necessary because control of the cooldown procedure is based on measurement of reactor coolant temperature, whereas the limiting pressure is actually dependent on the material temperature at the tip of the assumed flaw. During cooldown, the 1/4T vessel location is at a higher temperature than the fluid adjacent to the vessel ID. This condition, of course, is not true for the steady-state situation. It follows that at any given reactor coolant temperature, the ΔT developed during cooldown results in a higher value of $K_{I\cancel{R}C}$ at the 1/4T location for finite cooldown rates than for steady-state operation. Furthermore, if conditions exist such that the increase in $K_{I\cancel{R}C}$ exceeds K_{It}, the calculated allowable pressure during cooldown will be greater than the steady-state value.

The above procedures are needed because there is no direct control on temperature at the 1/4T location; therefore, allowable pressures may unknowingly be violated if the rate of cooling is decreased at various intervals along a cooldown ramp. The use of the composite curve eliminates this problem and assures conservative operation of the system for the entire cooldown period.

HEATUP

Three separate calculations are required to determine the limit curves for finite heatup rates. As is done in the cooldown analysis, allowable pressure-temperature relationships are developed for steady-state conditions as well as finite heatup rate conditions assuming the presence of a 1/4T defect at the inside of the vessel wall. The ~~thermal gradients during~~ heatup results in ~~produce~~ compressive stresses at the inside surface of the wall that alleviate the tensile stresses produced by internal pressure. The metal temperature at the crack tip lags the coolant temperature; therefore, the $K_{I\cancel{R}C}$ for the 1/4T crack during heatup is lower than the $K_{I\cancel{R}C}$ for the 1/4T crack during steady-state

3/4.4.9 PRESSURE/TEMPERATURE LIMITS (Continued)

HEATUP (Continued) lower C values

conditions at the same coolant temperature. During heatup, especially at the end of the transient, conditions may exist such that the effects of compressive thermal stresses and ~~different~~ K_{IR}~~s~~ 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.

flaw located at the ¼T location from the outside

The second portion of the heatup analysis concerns the calculation of pressure-temperature limitations for the case in which a 1/4T ~~deep outside~~ surface ~~flaw~~ 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, ~~of course,~~ are dependent on both the rate of heatup and the time (or coolant temperature) along the heatup ramp. ~~Furthermore,~~ 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 ~~such that~~ 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.

Insert A

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.

Insert B

Insert A, Page B3/4 4-14

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

Insert B, Page B3/4 4-14

References

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

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

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

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

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

6. Duke Engineering and Services Report DES-NFQA-98-01, Revision 0, "Analysis of Seabrook Station Unit 1 Reactor Vessel Surveillance Capsules U and Y", dated May 1998.

3/4.4.9 PRESSURE/TEMPERATURE LIMITS (Continued)

COLD OVERPRESSURE PROTECTION (Continued)

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 ~~329~~°F. 290 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

3/4.4.9 PRESSURE/TEMPERATURE LIMITS (Continued)

COLD OVERPRESSURE PROTECTION (Continued)

ESF actuation circuitry will result in the starting of at most one train of Safety Injection (one centrifugal charging pump, and one Safety Injection pump). For temperatures above 325°F, an overpressure event occurring as a result of starting two pumps can be successfully mitigated by operation of both PORVs without exceeding Appendix G limit. A single failure of a PORV is not assumed due to the short duration that this condition is allowed and the low probability of an event occurring during this interval in conjunction with the failure of a PORV to open. Initiation of both trains of Safety Injection during this 4-hour time frame due to operator error or a single failure occurring during testing of a redundant channel are not considered to be credible accidents.

Operation with all centrifugal charging pumps and both Safety Injection pumps OPERABLE is acceptable when RCS temperature is greater than 350°F, a single PORV has sufficient capacity to relieve the combined flow rate of all pumps. Above 350°F two RCPs and all pressure safety valves are required to be OPERABLE. Operation of an RCP eliminates the possibility of a 50°F difference existing between indicated and actual RCS temperature as a result of heat transport effects. Considering instrument uncertainties only, an indicated RCS temperature of 350°F is sufficiently high to allow full RCS pressurization in accordance with Appendix G limitations. Should an overpressure event occur in these conditions, the pressurizer safety valves provide acceptable and redundant overpressure protection.

When operating below 200°F in MODE 5 or MODE 6 with the reactor vessel head on and the vessel head closure bolts not fully detensioned, Technical Specification 3.5.3.2 allows one Safety Injection pump to be made OPERABLE whenever the RCS has a vent area equal to or greater than 18 square inches or whenever the RCS is in a reduced inventory condition, i.e., whenever reactor vessel water level is lower than 36 inches below the reactor vessel flange. Cold overpressure protection provided by the venting method utilizes an 18 square inch or greater mechanical opening in the RCS pressure boundary. This mechanical opening is larger in size than the 1.58 square inch opening required for normal overpressure protection and is of sufficient size to ensure that the Appendix G limits are not exceeded when an SI pump is operating in MODE 5 or MODE 6 with the reactor vessel head on and the vessel head closure bolts not fully detensioned. When the reactor has been shut down for at least 7 days, the larger vent area also enhances the ability to provide a gravity feed to the RCS from the Refueling Water Storage Tank in the unlikely event that the CCP and SI pumps were unavailable after a loss of RHR. Additionally, when steam generator nozzle dams are installed for maintenance purposes and the reactor vessel water level is not in a reduced inventory condition, the larger vent area limits RCS pressure during overpressure transients to reduce the possibility of adversely affecting steam generator nozzle dams.

For Information Only

3/4.4.9 PRESSURE/TEMPERATURE LIMITS (Continued)

COLD OVERPRESSURE PROTECTION (Continued)

When the reactor vessel head is on and the vessel head closure bolts are fully detensioned, i.e., when the closure nuts have been removed from the studs, a substantial vent area exists by the gap underneath the reactor vessel head, created by the internal spring forces. A measured gap of greater than or equal to 0.03 inches is of sufficient size to provide for cold overpressure protection, for gravity feed from the RWST, and ensuring nozzle dam integrity. Verification of sufficient gap will be performed prior to crediting the gap as a means for cold overpressure protection.

Cold overpressure protection can also be provided when operating at a reduced inventory condition, i.e., whenever reactor vessel water level is lower than 36 inches below the reactor vessel flange. With RCS water level lower than 36 inches below the RV flange in Mode 5 or Mode 6 with the RV head on and the closure bolts not fully detensioned, a mass addition transient involving simultaneous operation of a CCP and a SI pump without letdown will not result in a cold overpressurization condition because of the relatively large void volume in the RCS. This void volume consists of the upper plenum of the reactor vessel and the RV head, the pressurizer and steam generator tubes, as a minimum. The relatively large void volume affords ample time for operator action, (e.g., diagnose the water level increase on main control board instrumentation and stopping the pumps) to mitigate the transient. A minimum time of 50 minutes has been determined based on one charging pump operating at 120 gpm without letdown and a Safety Injection pump injecting into the RCS.

The charging pumps and Safety Injection pumps are rendered incapable of injecting into the RCS through removing the power from the pumps by racking the motor circuit breakers out under administrative control. An alternate method of preventing cold overpressurization may be employed. The alternate method uses at least two independent means to prevent cold overpressurization such that a single action will not result in an inadvertent injection into the RCS. This may be accomplished through the pump control switch being placed in Pull-to-Lock position and at least one valve in the discharge flow path closed. The alternate method provides the ability to respond to abnormal situations, expeditiously, from the main control room.

During charging pump swap operation two charging pumps may be made capable of injecting into the RCS for up to 1 hour. This provision prevents securing charging for the purpose of not having more than the allowable pumps operable in order to limit thermal fatigue cycles on piping and impact seal injection to the Reactor Coolant Pumps (RCP) which has seal degradation potential. Given the short time duration of the evolution and the evolution controlled under administrative controls, e.g., prohibiting pump swap operation during RCS water-solid conditions, a cold overpressurization condition occurring as a result of an uncontrolled mass addition transient is unlikely.

For Information Only

3/4.4.9 PRESSURE/TEMPERATURE LIMITS (Continued)

COLD OVERPRESSURE PROTECTION (Continued)

Charging and/or Safety Injection pumps, normally rendered inoperable for cold overpressure protection may be operated as required under administrative controls during abnormal situations involving a loss of decay heat removal capability or an unexpected reduction in RCS inventory. Maintaining adequate core cooling and RCS inventory during these abnormal situations is essential for public health and safety. Administrative controls ensure that a cold overpressurization condition will not occur as a result of an uncontrolled mass addition transient.

The Maximum Allowed PORV Setpoint for the Cold Overpressure Mitigation System will be revised on the basis of the results of examinations of reactor vessel material irradiation surveillance specimens performed as required by 10 CFR Part 50, Appendix H.

SECTION III

RETYPE OF PROPOSED CHANGES

Refer to the attached retype of the proposed changes to the Technical Specifications. The attached retype reflects the currently issued version of the Technical Specifications. Pending Technical Specification changes or Technical Specification changes issued subsequent to this submittal are not reflected in the enclosed retype. The enclosed retype should be checked for continuity with Technical Specifications prior to issuance.

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

3.0/4.0 BASES

MATERIAL PROPERTY BASIS

Limiting material: LOWER SHELL PLATE R-1808-1

Limiting ART values at 20 EFPY:1/4T, 109°F

3/4T, 88°F

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

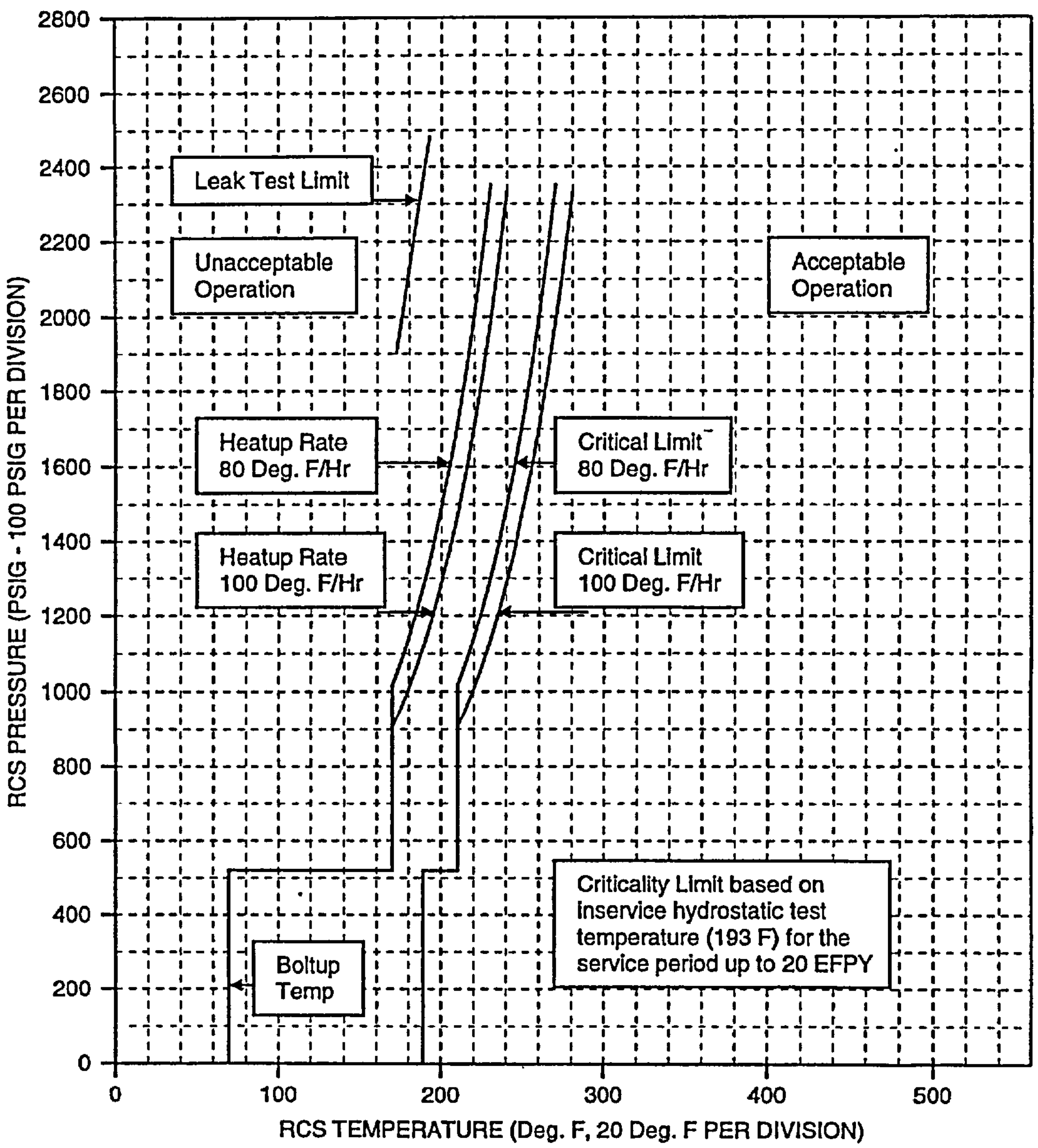


FIGURE 3.4-2

REACTOR COOLANT SYSTEM HEATUP LIMITATIONS - APPLICABLE UP TO 20 EFPY

MATERIAL PROPERTY BASIS

Limiting material: LOWER SHELL PLATE R-1808-1
Limiting ART values at 20 EFPY:1/4T, 109°F
3/4T, 88°F

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

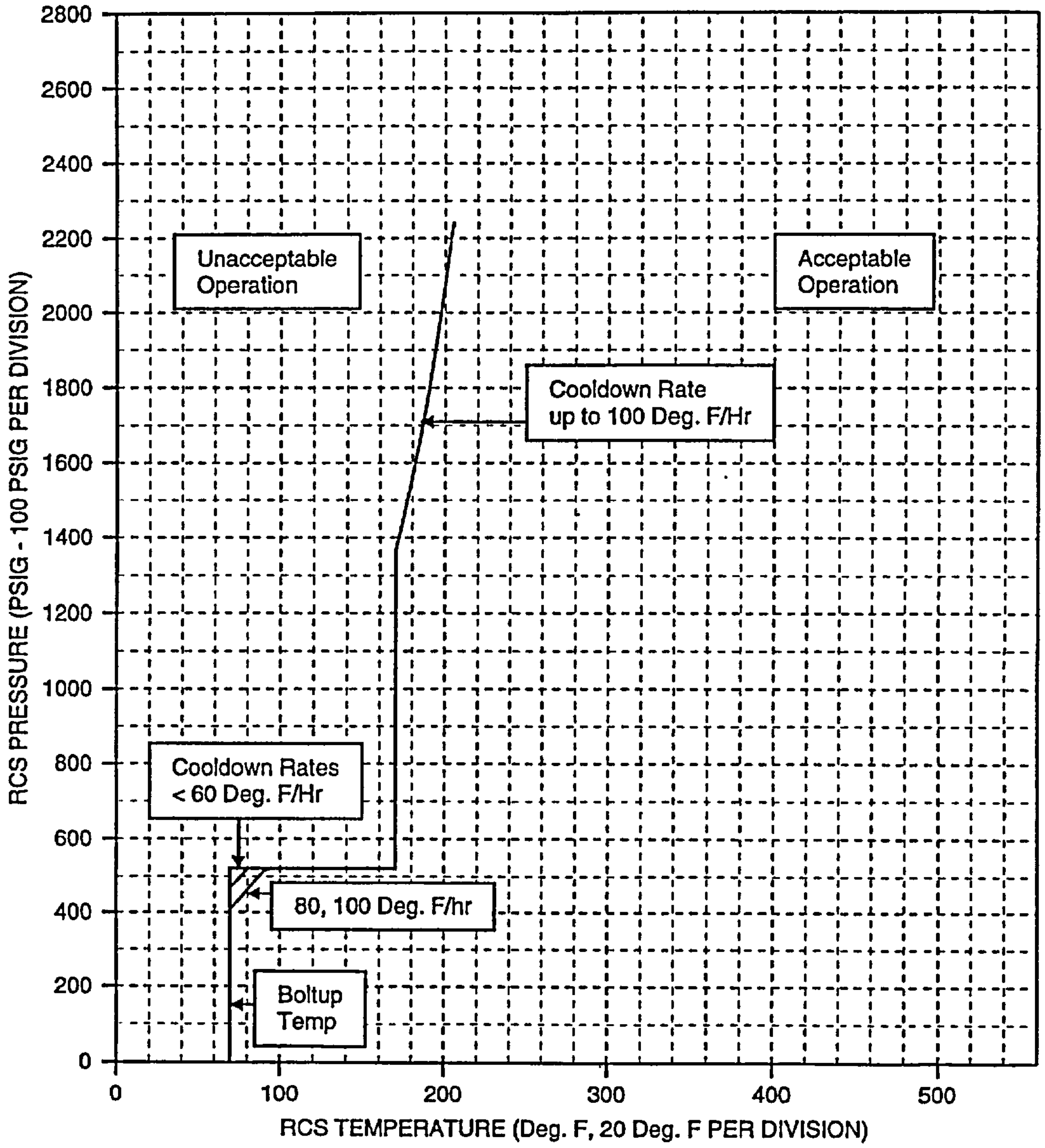


FIGURE 3.4-3
REACTOR COOLANT SYSTEM COOLDOWN LIMITATIONS - APPLICABLE UP TO 20 EFPY

LIMITING CONDITION FOR OPERATION

3.4.9.3 The following Overpressure Protection Systems shall be OPERABLE:

a. In MODE 4 when the temperature of any 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*.

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.

ACTION:

a) In MODE 4 with all Safety Injection pumps inoperable and with one of the two required overpressure protection devices inoperable, either restore two overpressure protection devices to OPERABLE status within 7 days or within the next 8 hours

(a) depressurize the RCS and

(b) vent the RCS through at least a 1.58-square-inch vent.

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

VALID FOR THE FIRST 20 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

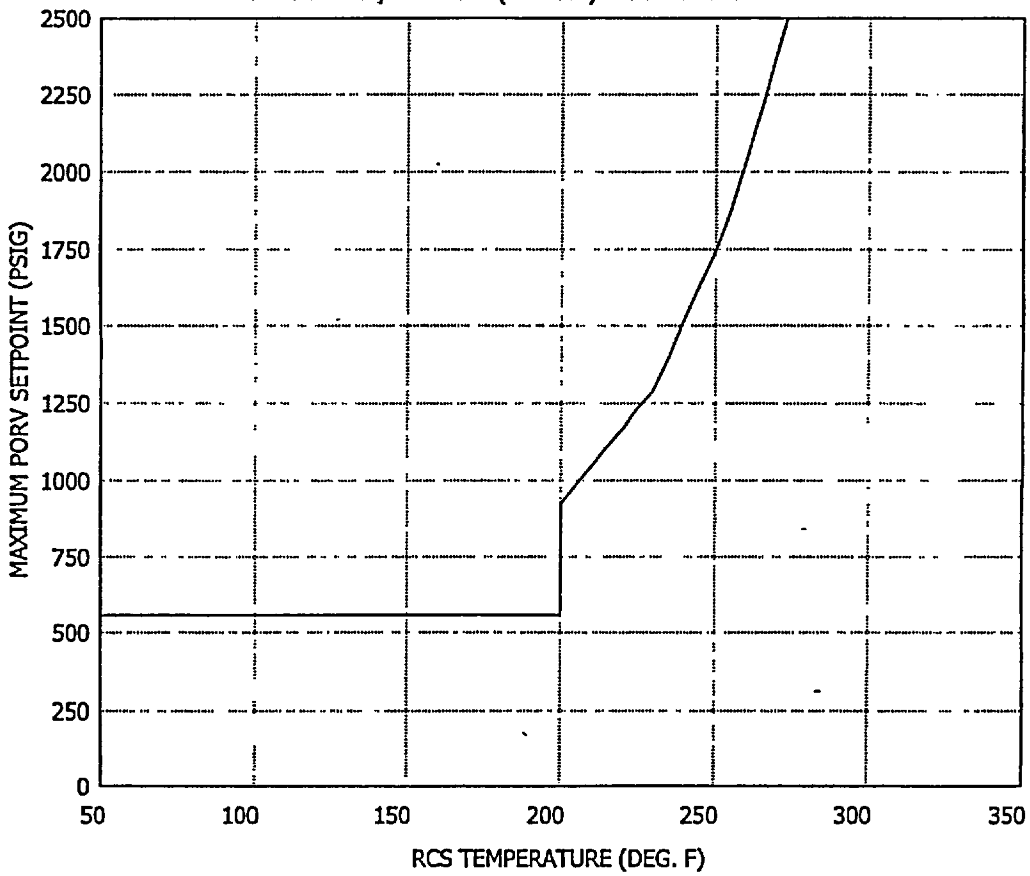


FIGURE 3.4-4 RCS COLD OVERPRESSURE PROTECTION SETPOINTS

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.

3/4.4.9 PRESSURE/TEMPERATURE LIMITS (Continued)

The P/T limits have been established in accordance with the requirements of ASME Boiler and Pressure Vessel Code Section XI, Appendix G, as modified by ASME Code Case N-641, Reference (2), and the additional requirements of 10CFR50 Appendix G, Reference (3). The heatup and cooldown P/T limit curves for normal operation, Figures 3.4-2 and 3.4-3 respectively, are valid for a service period of 20 effective full power years (EFPY). The technical justification and methodologies utilized in their development are documented in WCAP-15745, Reference (4). The P/T curves were generated based on the latest available reactor vessel information and latest calculated fluences.

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

The reactor vessel materials have been tested to determine their initial RT_{NDT}. Reactor operation and resultant fast neutron (E greater than 1 MeV) irradiation can cause an increase in the RT_{NDT}. Therefore, an adjusted reference temperature, based upon the fluence, best estimate copper and nickel content of the limiting beltline material, can be predicted using surveillance capsule data and the value of ΔRT_{NDT} computed by Regulatory Guide 1.99, Revision 2. Surveillance capsule data, documented in Reference (6), is available for two capsules (Capsules U and Y) having already been removed from the reactor vessel. This surveillance capsule data was used to calculate chemistry factor (CF) values per Position 2.1 of Regulatory Guide 1.99, Revision 2. It also noted that Reference (6) concluded that all the surveillance data was credible as the beltline material was behaving as empirically predicted. The heatup and cooldown limit curves of Figures 3.4-2 and 3.4-3 include predicted adjustments for this shift in RT_{NDT} at the end of 20 EFPY as well as adjustments for possible errors in the pressure and temperature sensing instruments.

3/4.4.9 PRESSURE/TEMPERATURE LIMITS (Continued)

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 (6). 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.

FIGURE B 3/4.4-1

(THIS FIGURE NUMBER IS NOT USED)

TABLE B 3/4.4-1

(THIS TABLE NUMBER IS NOT USED)

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:

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

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:

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

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)

COOLDOWN

For the calculation of the allowable pressure versus coolant temperature during cooldown, the reference flaw of Appendix G to the ASME Code is assumed to exist at the inside of the vessel wall. During cooldown, the controlling location of the flaw is always at the inside of the wall because the thermal gradients produce tensile stresses at the inside, which increase with increasing cooldown rates. Allowable pressure-temperature relations are generated for both steady-state and finite cooldown rate situations. From these relations, composite limit curves are constructed for each cooldown rate of interest.

The use of the composite curve in the cooldown analysis is necessary because control of the cooldown procedure is based on measurement of reactor coolant temperature, whereas the limiting pressure is actually dependent on the material temperature at the tip of the assumed flaw. During cooldown, the 1/4T vessel location is at a higher temperature than the fluid adjacent to the vessel ID. This condition, of course, is not true for the steady-state situation. It follows that at any given reactor coolant temperature, the ΔT developed during cooldown results in a higher value of K_{IC} at the 1/4T location for finite cooldown rates than for steady-state operation. Furthermore, if conditions exist such that the increase in K_{IC} exceeds K_{It}, the calculated allowable pressure during cooldown will be greater than the steady-state value.

The above procedures are needed because there is no direct control on temperature at the 1/4T location; therefore, allowable pressures may unknowingly be violated if the rate of cooling is decreased at various intervals along a cooldown ramp. The use of the composite curve eliminates this problem and assures conservative operation of the system for the entire cooldown period.

HEATUP

Three separate calculations are required to determine the limit curves for finite heatup rates. As is done in the cooldown analysis, allowable pressure-temperature relationships are developed for steady-state conditions as well as finite heatup rate conditions assuming the presence of a 1/4T defect at the inside of the vessel wall. The heatup results in compressive stresses at the inside surface of the wall that alleviate the tensile stresses produced by internal pressure. The metal temperature at the crack tip lags the coolant temperature; therefore, the K_{IC} for the 1/4T crack during heatup is lower than the K_{IC} for the 1/4T crack during steady-state conditions at the same coolant temperature.

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)

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

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

References

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

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

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

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

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

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

3/4.4.9 PRESSURE/TEMPERATURE LIMITS (Continued)

COLD OVERPRESSURE PROTECTION (Continued)

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

SECTION IV

DETERMINATION OF SIGNIFICANT HAZARDS FOR PROPOSED CHANGES

IV. DETERMINATION OF SIGNIFICANT HAZARDS FOR PROPOSED CHANGES

License Amendment Request (LAR) 02-04 proposes changes to the Seabrook Station Technical Specifications (TSs) 3.4.9.1, "Reactor Coolant System – Pressure/Temperature Limits" and 3.4.9.3, "Reactor Coolant System – Overpressure Protection Systems". Specifically, the proposed changes will replace Technical Specification Figure 3.4-2, "Reactor Coolant System Heatup Limitations", Figure 3.4-3, "Reactor Coolant System Cooldown Limitations" and Figure 3.4-4, "RCS Cold Overpressure Protection Setpoints". The proposed change of the reactor vessel pressure-temperature (P/T) limit curves is required because the existing referenced curves are valid through the attainment of 11.1 Effective Full Power Years (EFPY). Based on current projections, 11.1 EFPY will be achieved early in the next operating cycle, Cycle No. 10, at the end of the fourth quarter of 2003. The revised reactor vessel P/T limit curves will remain valid for 20 EFPY as demonstrated in the analysis documented in Reference (1). The P/T limit curves were generated using the most recent reactor vessel surveillance capsule data, which is documented in Reference (3).

An evaluation of the proposed change has been performed in accordance with 10CFR50.91(a)(1) regarding no significant hazards considerations using the criteria in 10CFR50.92(c). A discussion of these criteria as they relate to this amendment request follows:

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

   The proposed changes to TS 3.4.9.1 and TS 3.4.9.3 do not result in a condition where the design, material, and construction standards that were applicable prior to the proposed changes are altered. The probability of occurrence of an accident previously evaluated for Seabrook Station is not altered by the proposed amendment to the TSs. The accidents remain the same as currently analyzed in the UFSAR as a result of changes to the P/T limits as well as those for Cold Overpressure Mitigation System (COMS). The new P/T limits are based on NRC accepted methodology along with American Society of Mechanical Engineers (ASME) Code alternative methodology. An exemption request to allow use of the alternative ASME methodology is included as part of this LAR. The proposed COMS setpoint limit based on the revised P/T limits satisfies the criteria specified in the alternative ASME methodology and 10 CFR Part 50 Appendix G closure head/vessel flange region pressure limit criteria. The proposed changes do not impact the integrity of the reactor coolant pressure boundary (RCPB) i.e. there is no change to the operating pressure, materials, system loadings, etc., as a result of this change. In addition, there is no increase in the potential for the occurrence of a loss of coolant accident. The probability of any design basis accident is not affected by this change, nor are the consequences of any design basis accident (DBA) affected by this proposed change. The proposed P/T limit curves and the COMS limits are not considered to be an initiator or contributor to any accident currently, evaluated in the Seabrook Station UFSAR. These new limits ensure the long term structural integrity of the RCPB.

   Fracture toughness test data are obtained from beltline material specimens contained in surveillance capsules that are periodically withdrawn from the reactor vessel. This data allows determination of time conditions under which the vessel can be operated with adequate safety margins against non-ductile fracture throughout its service life. The second Seabrook Station surveillance capsule was removed from the reactor vessel after completion of Operating Cycle No. 5 in May 1997 and was analyzed to predict the fracture toughness requirements using projected neutron fluence calculations. For each analyzed transient and steady state condition, the allowable pressure is determined as a function of reactor coolant temperature considering postulated flaws in the reactor vessel beltline region material. The predicted radiation induced ΔRT_{NDT} was calculated using the respective reactor vessel beltline materials copper and nickel contents and the neutron fluence predicted for 20 EFPY. The RT_{NDT} and, accordingly, the operating limits for Seabrook Station were adjusted to account for the effects of irradiation on the fracture toughness of the reactor vessel beltline materials. Therefore, new operating limits are established which are represented in the revised operating curves for heatup/cooldown, criticality and inservice hydrostatic testing contained in the technical specifications. The proposed P/T limit curves and COMS setpoint limits are not considered to be an initiator or contributor to any accident currently evaluated in the Seabrook Station UFSAR.

Therefore based on the above discussion, it is concluded that the proposed revisions to TS 3.4.9.1 and TS 3.4.9.3 do not involve a significant increase in the probability or consequences of any accident previously evaluated.

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

   The proposed changes to the P/T and COMS limits will not create a new accident scenario. The requirements to have P/T and COMS protection are part of the licensing basis for Seabrook Station. The proposed technical specification amendment reflects the change in reactor vessel material properties as determined by evaluation of the most recently withdrawn surveillance capsule. Based on the surveillance capsule data, the adjusted RT_{NDT} values for the plate and weld material were within the two standard deviations of Regulatory Guide 1.99, Revision 2 predictions. As all the requisite criteria of Regulatory Guide 1.99, Revision 2 was satisfied, it was concluded that the surveillance data was credible and the beltline material was responding as empirically predicted. The new P/T limits are based on NRC accepted methodology along with American Society of Mechanical Engineers (ASME) Code alternative methodology. An exemption request to allow use of the alternative ASME methodology is included as part of this LAR. The proposed COMS setpoint limit based on the revised P/T limits satisfies the criteria specified in the alternative ASME methodology and 10 CFR Part 50 Appendix G closure head/vessel flange region pressure limit criteria. The proposed changes will not alter the way any structure, system or component functions, and will not significantly alter the manner in which the plant is operated. There will be no adverse effect on plant operation or accident mitigation equipment.

Since no new failure modes are created by the proposed revisions to TS 3.4.9.1 and TS 3.4.9.3, this change does not create the possibility of a new or different kind of accident from any that was previously evaluated.

3. The proposed changes do not involve a significant reduction in the margin of safety.

The existing P/T and COMS limit curves in the technical specifications are reaching their expiration for the number of years at effective full power operation. The revision of the P/T limits and COMS will ensure that Seabrook Station continues to operate within the operating limits allowed by 10CFR50.60 and the ASME Code. The material properties used in the development of the revised limit curves are based on the evaluation of the most recently withdrawn surveillance capsule. The application of ASME Code Case N-641 presents alternative methods for calculating P/T and COMS temperature and pressure limits in lieu of those established in ASME Section XI, Appendix G-2215. This ASME Code alternative allows analysis features that are less restrictive than those associated with previous methodologies, however these features remain conservative with respect to the requirements delineated ASME Section XI. Therefore it is concluded that the revised P/T and COMS limit curves proposed by this technical specification amendment still provide sufficient margin to preclude non-ductile fracture of the reactor vessel.

Thus, it is concluded that these proposed revisions to TS 3.4.9.1 and TS 3.4.9.3 do not involve a significant reduction in a margin of safety.

Therefore, based upon the evaluation presented above and the previous discussion of the amendment request, North Atlantic concludes that the proposed revisions to TS 3.4.9.1 and TS 3.4.9.3 do not constitute a significant hazard as defined by the criteria in 10CFR50.92(c).

SECTIONS V AND VI

PROPOSED SCHEDULE FOR LICENSE AMENDMENT ISSUANCE AND EFFECTIVENESS

AND

ENVIRONMENTAL IMPACT ASSESSMENT

V. PROPOSED SCHEDULE FOR LICENSE AMENDMENT ISSUANCE AND EFFECTIVENESS

North Atlantic requests NRC review of License Amendment Request 02-04, and issuance of a license amendment by September 30, 2003, having immediate effectiveness and implementation within 60 days. Issuance of a license amendment by the requested date would afford North Atlantic the flexibility for planning of technical resources in support of Seabrook Station's Tenth Operating Cycle scheduled to commence in the fourth quarter of 2003.

VI. ENVIRONMENTAL IMPACT ASSESSMENT

North Atlantic has reviewed the proposed license amendment against the criteria of 10CFR51.22 for environmental considerations. 10CFR51.22(c) provides criteria for and identification of licensing and regulatory actions eligible categorical exclusion from performing an environmental assessment. A proposed amendment to an operating license for a facility requires no environmental assessment if operation of the facility in accordance with the proposed amendment would not: (1) involve a significant hazard consideration, (2) result in a significant change in the types or significant increase in the amounts of any effluents that may be released offsite or (3) result in a significant increase in individual or cumulative occupational radiation exposure. North Atlantic has reviewed this proposed license amendment and has determined that it meets the eligibility criteria for categorical exclusion set forth in 10CFR51.22(c)(9). Pursuant to 10CFR51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the proposed license amendment.
The basis for this determination is as follows:

1. The proposed license amendment does not involve a significant hazards consideration described previously in this evaluation (Section IV).

2. As discussed in the significant hazards evaluation, the proposed amendment does not result in a significant change or significant increase in the radiological doses for any Design Basis Accident. The proposed license amendment does not result in a significant change in the types or a significant increase in the amounts of any effluents that may be released off-site.

3. The proposed license amendment does not result in a significant increase to the individual or cumulative occupational radiation exposure because the proposed update of the operating P/T and COMS limits does not impact the exposure of plant personnel.

Based on the preceding discussion, North Atlantic concludes that the proposed changes meet the criterion delineated in 10 CFR 51.22(c)(9) for a categorical exclusion from the requirements for an Environmental Impact Statement.

ENCLOSURE 2 TO NYN-02093

Westinghouse Non-Proprietary Class 3

WCAP-15745

Seabrook Unit 1 Heatup and Cooldown Limit Curves for Normal Operation

Westinghouse Electric Company LLC

Cover

WCAP-15745

Seabrook Unit 1 Heatup and Cooldown Limit Curves for Normal Operation

T. J. Laubham

December 2001

Prepared by the Westinghouse Electric Company LLC
for the North Atlantic Energy Services Corporation

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PREFACE

This report has been technically reviewed and verified by:

J.H. Ledger ____________

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EXECUTIVE SUMMARY

This report provides the methodology and results of the generation of heatup and cooldown pressure temperature limit curves for normal operation of the Seabrook Unit 1 reactor vessel. The PT curves were generated based on the latest available reactor vessel information and latest calculated fluences. The new Seabrook Unit 1 heatup and cooldown pressure-temperature limit curves were generated using ASME Code Case N-641[3] (which allows the use of the K_{Ic} methodology) and the axial flaw methodology of the 1995 ASME Code, Section XI through the 1996 Addenda. It should be noted that the Seabrook reactor vessel was limited by the lower shell plate R-1808-1. The pressure-temperature (PT) limit curves and data points are presented in Section 5.

1 INTRODUCTION

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. The unirradiated RT_{NDT} is designated as the higher of either the drop weight nil-ductility transition temperature (NDTT) or the temperature at which the material exhibits at least 50 ft-lb of impact energy and 35-mil lateral expansion (normal to the major working direction) minus 60°F.

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, "Radiation Embrittlement of Reactor Vessel Materials."[1] 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 measured from the clad/base metal interface.

The heatup and cooldown curves documented in this report were generated using the most limiting ART values and the NRC approved methodology documented in WCAP-14040-NP-A, Revision 2[2], "Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves" with exception of the following: 1) **The fluence values used in this report are calculated fluence values, not the best estimate fluence values.** 2) The K_{Ic} critical stress intensities are used in place of the K_{Ia} critical stress intensities. This methodology is taken from approved ASME Code Case N-641[3]. 3) The 1996 Version of Appendix G to Section XI[4] will be used rather than the 1989 version.

2 FRACTURE TOUGHNESS PROPERTIES

The fracture-toughness properties of the ferritic materials in the reactor coolant pressure boundary are determined in accordance with the NRC Standard Review Plan[5]. The beltline material properties of the Seabrook Unit 1 reactor vessel is presented in Table 1.

Best estimate copper (Cu) and nickel (Ni) weight percent values used to calculate chemistry factors (CF) in accordance with Regulatory Guide 1.99, Revision 2, are provided in Table 1. Additionally, surveillance capsule data is available for two capsules (Capsules U and Y) already removed from the Seabrook Unit 1 reactor vessel. This surveillance capsule data was also used to calculate CF values per Position 2.1 of Regulatory Guide 1.99, Revision 2 in Table 2. These CF values are summarized in Table 3. It should be noted that all the surveillance data has been determined to be credible.

The Regulatory Guide 1.99, Revision 2 methodology used to develop the heatup and cooldown curves documented in this report is the same as that documented in WCAP-14040, Revision 2.

TABLE 1

Summary of the Best Estimate Cu and Ni Weight Percent and Initial RT_{NDT} Values for the Seabrook Unit 1 Reactor Vessel Materials

Material Description	Cu (%)	Ni(%)	Initial RT_{NDT}[a]
Closure Head Flange	---	0.76	10°F
Vessel Flange	---	0.73	30°F
Intermediate Shell Plate R-1806-1[d]	0.045	0.61	40°F
Intermediate Shell Plate R-1806-2[d]	0.06	0.64	0°F
Intermediate Shell Plate R-1806-3[d]	0.075	0.63	10°F
Lower Shell Plate R-1808-1[d]	0.06	0.58	40°F
Lower Shell Plate R-1808-2[d]	0.06	0.58	10°F
Lower Shell Plate R-1808-3[d]	0.07	0.59	40°F
Beltline Weld Seams (Heat # 4P6052)[a]	0.047	0.049	-60°F[e]
Seabrook Unit 1 Surveillance Weld (Heat # 4P6052)[c]	0.02	0.075	---

Notes:

(a) The Beltline Weld Seams Consist of the Intermediate Shell Longitudinal Welds (101-124A,B,C), the Lower Shell Longitudinal Welds (101-142A,B,C) and the Intermediate to Lower Shell Girth Weld (101-171). These welds were fabricated with Wire Heat No. 4P6052, Flux Type 0091, Flux Lot No. 0145. The copper and Nickel weight percents were taken from CE Reports NPSD-1039, Rev. 2[6] & NPSD-1119, Rev. 1[7]. It should be noted that these Cu & Ni values do not Match RVID2 however, they would produce a more conservative Table Chemistry factor versus those in RVID2. This fact is negligible since the welds are not limiting.

(b) The Initial RT_{NDT} values are measured values unless otherwise noted.

(c) Average of the two data points presented in Table A-3 of WCAP-10110[8].

(d) Average of Lukens Mill Test Report and CE Test (Documented in WCAP-10110).

(e) Measured value documented in WCAP-10110

The chemistry factors were calculated using Regulatory Guide 1.99 Revision 2, Positions 1.1 and 2.1. Position 1.1 uses the Tables from the Reg. Guide along with the best estimate copper and nickel weight percents. Position 2.1 uses the surveillance capsule data from all capsules withdrawn to date. The fluence values used to determine the CFs in Table 3 are the calculated fluence values at the surveillance capsule locations. Hence, the calculated fluence values were used for all cases. All calculated fluence values (capsule and projections) for Seabrook Unit 1 were documented in DES-NFQA-98-01[9]. 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.

TABLE 2

Calculated Integrated Neutron Exposure of the Surveillance Capsules @ Seabrook Unit 1

Capsule	Fluence
Seabrook Unit 1[a]	
U	2.55×10^{18} n/cm^2, (E > 1.0 MeV)
Y	1.031×10^{19} n/cm^2, (E > 1.0 MeV)

NOTES:

(a) Per Table 6-12 of DES-NFQA-98-01[9].

TABLE 3

Calculation of Chemistry Factors using Seabrook Unit 1 Surveillance Capsule Data

Material	Capsule	Capsule f[a]	FF[b]	ΔRT_{NDT}[c]	FF*ΔRT_{NDT}	FF^2
Lower Shell Plate R-1808-3 (Long.)	U	0.255	0.629	36.0	22.644	0.396
	Y	1.031	1.009	44.0	44.396	1.018
Lower Shell Plate R-1808-3 (Trans.)	U	0.255	0.629	28.0	17.612	0.396
	Y	1.031	1.009	34.0	34.306	1.018
				SUM:	118.958	2.828
	$CF_{B\text{-}2002\text{-}2} = \Sigma(FF * RT_{NDT}) \div \Sigma(FF^2) = (118.958) \div (2.828) = 42.1°F$					
Surveillance Weld Material[d]	U	0.255	0.629	11.4 (10.0)	7.171	0.396
	Y	1.031	1.009	11.4 (10.0)	11.503	1.018
				SUM:	18.674	1.414
	$CF_{Surv.\ Weld} = \Sigma(FF * RT_{NDT}) \div \Sigma(FF^2) = (18.674°F) \div (1.414) = 13.2°F$					

Notes:

(a) f = fluence. See Table 2, (x 10^{19} n/cm^2, E > 1.0 MeV).

(b) FF = fluence factor = $f^{(0.28 - 0.1*\log f)}$.

(c) ΔRT_{NDT} values are the measured 30 ft-lb shift values taken from DES-NFQA-98-01[9].

(d) The surveillance weld metal ΔRT_{NDT} values have been adjusted by a ratio factor 1.14. The pre-adjusted values are in parenthesis.

TABLE 4

Summary of the Seabrook Unit 1 Reactor Vessel Beltline Material Chemistry Factors

Material	Reg. Guide 1.99, Rev. 2 Position 1.1 CF's	Reg. Guide 1.99, Rev. 2 Position 2.1 CF's
Intermediate Shell Plate R-1806-1	28.5°F	---
Intermediate Shell Plate R-1806-2	37°F	---
Intermediate Shell Plate R-1806-3	47.5°F	---
Lower Shell Plate R-1808-1	37°F	---
Lower Shell Plate R-1808-2	37°F	---
Lower Shell Plate R-1808-3	44°F	42.1°F
Intermediate & Lower Shell Longitudinal Weld Seams (Heat # 4P6052)	30.7°F	13.2°F
Intermediate to Lower Shell Girth Weld Seam (Heat # 4P6052)	30.7°F	13.2°F
Seabrook Unit 1 Surveillance Weld (Heat # 4P6052)	27°F	---

3 CRITERIA FOR ALLOWABLE PRESSURE-TEMPERATURE RELATIONSHIPS

3.1 Overall Approach

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, "Alternative Pressure-Temperature Relationship and Low Temperature Overpressure Protection System Requirements Section XI, Division 1"[3 & 4] of the ASME Appendix G to Section XI. The K_{Ic} curve is given by the following equation:

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

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 Class1, SA-508-1, SA-508-2, SA-508-3 steel.

3.2 Methodology for Pressure-Temperature Limit Curve Development

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

$$C^* K_{Im} + K_{It} < K_{Ic} \tag{2}$$

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

For membrane tension, the corresponding K_I for the postulated defect is:

$$K_{Im} = M_m \times (pR_i / t) \tag{3}$$

where, M_m for an inside surface flaw is given by:

M_m = 1.85 for $\sqrt{t} < 2$,
M_m = $0.926\sqrt{t}$ for $2 \le \sqrt{t} \le 3.464$,
M_m = 3.21 for $\sqrt{t} > 3.464$

Similarly, M_m for an outside surface flaw is given by:

M_m = 1.77 for $\sqrt{t} < 2$,
M_m = $0.893\sqrt{t}$ for $2 \le \sqrt{t} \le 3.464$,
M_m = 3.09 for $\sqrt{t} > 3.464$

and p = internal pressure, Ri = vessel inner radius, and t = vessel wall thickness.

For bending stress, the corresponding K_I for the postulated defect is:

K_{Ib} = M_b * Maximum Stress, where M_b is two-thirds of M_m

The maximum K_I produced by radial thermal gradient for the postulated inside surface defect of G-2120 is $K_{It} = 0.953 \times 10^{-3} \times CR \times t^{2.5}$, where CR is the cooldown rate in °F/hr., or for a postulated outside surface defect, $K_{It} = 0.753 \times 10^{-3} \times HU \times t^{2.5}$, where HU is the heatup rate in °F/hr.

The through-wall temperature difference associated with the maximum thermal K_I can be determined from Fig. G-2214-1. The temperature at any radial distance from the vessel surface can be determined from Fig. G-2214-2 for the maximum thermal K_I.

(a) The maximum thermal K_I relationship and the temperature relationship in Fig. G-2214-1 are applicable only for the conditions given in G-2214.3(a)(1) and (2).

(b) Alternatively, the K_I for radial thermal gradient can be calculated for any thermal stress distribution and at any specified time during cooldown for a ¼-thickness inside surface defect using the relationship:

$$K_{It} = (1.0359C_0 + 0.6322C_1 + 0.4753C_2 + 0.3855C_3) * \sqrt{\pi a} \tag{4}$$

or similarly, K_{IT} during heatup for a ¼-thickness outside surface defect using the relationship:

$$K_{It} = (1.043C_0 + 0.630C_1 + 0.481C_2 + 0.401C_3) * \sqrt{\pi a} \quad (5)$$

where the coefficients C_0, C_1, C_2 and C_3 are determined from the thermal stress distribution at any specified time during the heatup or cooldown using the form:

$$\sigma(x) = C_0 + C_1(x/a) + C_2(x/a)^2 + C_3(x/a)^3 \quad (6)$$

and x is a variable that represents the radial distance from the appropriate (i.e., inside or outside) surface to any point on the crack front and a is the maximum crack depth.

Note, that equations 3, 4 and 5 were implemented in the OPERLIM computer code, which is the program used to generate the pressure-temperature (P-T) limit curves. No other changes were made to the OPERLIM computer code with regard to P-T calculation methodology. Therefore, the P-T curve methodology is unchanged from that described in WCAP-14040, "Methodology used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves"[2] Section 2.6 (equations 2.6.2-4 and 2.6.3-1) with the exceptions just described above.

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.

For the calculation of the allowable pressure versus coolant temperature during cooldown, the reference flaw of Appendix G to the ASME Code is assumed to exist at the inside of the vessel wall. During cooldown, the controlling location of the flaw is always at the inside of the wall because the thermal gradients produce tensile stresses at the inside, which increase with increasing cooldown rates. Allowable pressure-temperature relations are generated for both steady-state and finite cooldown rate situations. From these relations, composite limit curves are constructed for each cooldown rate of interest.

The use of the composite curve in the cooldown analysis is necessary because control of the cooldown procedure is based on the measurement of reactor coolant temperature, whereas the limiting pressure is actually dependent on the material temperature at the tip of the assumed flaw. During cooldown, the 1/4T vessel location is at a higher temperature than the fluid adjacent to the vessel inner diameter. This condition, of course, is not true for the steady-state situation. It follows that, at any given reactor coolant temperature, the ΔT (temperature) developed during cooldown results in a higher value of K_{Ic} at the 1/4T location for finite cooldown rates than for steady-state operation. Furthermore, if conditions exist so that the increase in K_{Ic} exceeds K_{It}, the calculated allowable pressure during cooldown will be greater than the steady-state value.

The above procedures are needed because there is no direct control on temperature at the 1/4T location and, therefore, allowable pressures may unknowingly be violated if the rate of cooling is decreased at various intervals along a cooldown ramp. The use of the composite curve eliminates this problem and ensures conservative operation of the system for the entire cooldown period.

Three separate calculations are required to determine the limit curves for finite heatup rates. As is done in the cooldown analysis, allowable pressure-temperature relationships are developed for steady-state conditions as well as finite heatup rate conditions assuming the presence of a 1/4T defect at the inside of the wall. The heatup results in compressive stresses at the inside surface that alleviate the tensile stresses produced by internal pressure. The metal temperature at the crack tip lags the coolant temperature; therefore, the K_{Ic} for the 1/4T crack during heatup is lower than the K_{Ic} for the 1/4T crack during steady-state conditions at the same coolant temperature. During heatup, especially at the end of the transient, conditions may exist so that the effects of compressive thermal stresses and lower K_{Ic} values 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 ensure 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 the 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 therefore 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 rates, each heatup rate 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 by constructing a composite curve 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.

3.3 Closure Head/Vessel Flange Requirements

10 CFR Part 50, Appendix G[10] 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 is 621 psig for Seabrook Unit 1. The limiting unirradiated RT_{NDT} of 30°F occurs in the vessel flange of the Seabrook Unit 1 reactor vessel, so the minimum allowable temperature of this region is 150°F at pressures greater than 621 psig. This limit is shown in Figures 1 through 4 wherever applicable.

3.4 LTOP System Allowable Pressure

Per Code Case N-641[3], the LTOP system shall limit the maximum pressure in the vessel to 100% of the pressure determined by Equation 2, herein, if K_{Ic} is used for determining allowable pressure, which is the case for this report.

4 CALCULATION OF ADJUSTED REFERENCE TEMPERATURE

From Regulatory Guide 1.99, Revision 2, the adjusted reference temperature (ART) for each material in the beltline region is given by the following expression:

$$ART = \text{Initial } RT_{NDT} + \Delta RT_{NDT} + \text{Margin} \tag{7}$$

Initial RT_{NDT} is the reference temperature for the unirradiated material as defined in paragraph NB-2331 of Section III of the ASME Boiler and Pressure Vessel Code[11]. If measured values of initial RT_{NDT} for the material in question are not available, generic mean values for that class of material may be used if there are sufficient test results to establish a mean and standard deviation for the class.

ΔRT_{NDT} is the mean value of the adjustment in reference temperature caused by irradiation and should be calculated as follows:

$$\Delta RT_{NDT} = CF * f^{(0.28 - 0.10 \log f)} \tag{8}$$

To calculate ΔRT_{NDT} at any depth (e.g., at 1/4T or 3/4T), the following formula must first be used to attenuate the fluence at the specific depth.

$$f_{(depth\, x)} = f_{surface} * e^{(-0.24x)} \tag{9}$$

where x inches (vessel beltline thickness is 8.63 inches) is the depth into the vessel wall measured from the vessel clad/base metal interface. The resultant fluence is then placed in Equation 8 to calculate the ΔRT_{NDT} at the specific depth.

Duke Engineering & Services evaluated the vessel fluence projections in DES-NFQA-98-01[9] and are also presented in a condensed version in Table 5 of this report. The evaluation used the ENDF/B-VI scattering cross-section data set. This is consistent with methods presented in WCAP-14040-NP-A, "Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves"[2]. Table 5 contains the calculated vessel surface fluences values at various azimuthal locations. Tables 6 and 7 contain the 1/4T and 3/4T calculated fluences and fluence factors, per the Regulatory Guide 1.99, Revision 2, used to calculate the ART values for all beltline materials in the Seabrook Unit 1 reactor vessel.

TABLE 5

Calculated Neutron Fluence Projections[b] at Key Locations on the Reactor Vessel Clad/Base Metal Interface (10^{19} n/cm^2, E > 1.0 MeV)

EFPY	Azimuthal Location				
	0°	19°-21.5°	29°	31.5°	44°-45°
5.572[a]	0.201	0.355	0.196	0.197	0.369
16[c]	0.577	1.019	0.562	0.565	1.059
20[c]	0.722	1.273	0.702	0.706	1.324
32	1.155	2.037	1.123	1.130	2.119

Notes:

(a) Date of last capsule removal.

(b) Determined by multiplying Best Estimate Fluences by the Calculated-to-Measured Ratio (Ratio = 0.893).

(c) Values have been interpolated between 5.572 EFPY and 32 EFPY.

TABLE 6

Summary of the Vessel Surface, 1/4T and 3/4T Fluence Values used for the Generation of the 16 and 20 EFPY Heatup/Cooldown Curves

Material	Surface	¼ T[a]	¾ T[a]
16 EFPY			
Beltline Materials[b]	1.059×10^{19}	6.31×10^{18}	2.24×10^{18}
20 EFPY			
Beltline Materials[b]	1.324×10^{19}	7.89×10^{18}	2.80×10^{18}

Note:

(a) 1/4T and 3/4T = $F_{(Surface)} * e^{(-0.24*x)}$, where x is the depth into the vessel wall (i.e. 8.63*0.25 or 0.75)

(b) The beltline materials consist of the intermediate shell plates (R-1806-1,2,3), the lower shell plates (R-1808-1,2,3) intermediate and lower shell longitudinal welds and the intermediate to lower shell girth weld. Since the limiting material is a plate it would be subjected to the peak vessel fluence. Thus, for conservatism the peak vessel fluence was applied to the weld seams that are not at the peak azimuthal fluence location.

TABLE 7
Summary of the Calculated Fluence Factors used for the Generation of the 16 and 20 EFPY Heatup and Cooldown Curves

Material	1/4T F (n/cm^2, E > 1.0 MeV)	1/4T FF	3/4T F (n/cm^2, E > 1.0 MeV)	3/4T FF
16 EFPY				
Beltline Materials[a]	6.31×10^{18}	0.871	2.24×10^{18}	0.597
20 EFPY				
Beltline Materials[a]	7.89×10^{18}	0.934	2.80×10^{18}	0.653

Note:
(a) The beltline materials consist of the intermediate shell plates (R-1806-1,2,3), the lower shell plates (R-1808-1,2,3) intermediate and lower shell longitudinal welds and the intermediate to lower shell girth weld. Since the limiting material is a plate it would be subjected to the peak vessel fluence. Thus, for conservatism the peak vessel fluence was applied to the weld seams that are not at the peak azimuthal fluence location.

Margin is calculated as, $M = 2\sqrt{\sigma_i^2 + \sigma_\Delta^2}$. The standard deviation for the initial RT_{NDT} margin term, is σ_i 0°F when the initial RT_{NDT} is a measured value, and 17°F when a generic value is available. The standard deviation for the ΔRT_{NDT} margin term, σ_Δ, is 17°F for plates or forgings, and 8.5°F for plates or forgings when surveillance data is used. For welds, σ_Δ is equal to 28°F when surveillance capsule data is not used, and is 14°F (half the value) when credible surveillance capsule data is used. σ_Δ need not exceed 0.5 times the mean value of ΔRT_{NDT}.

Contained in Tables 8 through 11 are the calculations of the 16 and 20 EFPY ART values used for generation of the heatup and cooldown curves.

TABLE 8
Calculation of the ART Values for the 1/4T Location @ 16 EFPY

Material	RG 1.99 R2 Method	CF (°F)	FF	IRT_{NDT}[(a)] (°F)	ΔRT_{NDT}[(b)] (°F)	Margin[(c, e)] (°F)	ART[(d)] (°F)
Intermediate Shell Plate R-1806-1	Position 1.1	28.5	0.871	40	24.8	24.8	90
Intermediate Shell Plate R-1806-2	Position 1.1	37	0.871	0	32.2	32.2	64
Intermediate Shell Plate R-1806-3	Position 1.1	47.5	0.871	10	41.4	34	85
Lower Shell Plate R-1808-1	Position 1.1	37	0.871	40	32.2	32.2	**104**
Lower Shell Plate R-1808-2	Position 1.1	37	0.871	10	32.2	32.2	74
Lower Shell Plate R-1808-3	Position 1.1	44	0.871	40	38.3	34	112
	Position 2.1	42.1	0.871	40	36.7	17	94
Intermediate & Lower Shell	Position 1.1	30.7	0.871	-60	26.7	26.7	-7
Longitudinal Weld Seams (Heat # 4P6052)	Position 2.1	13.2	0.871	-60	11.5	11.5	-37
Inter. to Lower Shell Girth	Position 1.1	30.7	0.871	-60	26.7	26.7	-7
Weld Seam (Heat # 4P6052)	Position 2.1	13.2	0.871	-60	11.5	11.5	-37

Notes:
(a) Initial RT_{NDT} values are measured values except for the welds.
(b) $\Delta RT_{NDT} = CF * FF$
(c) $M = 2 * (\sigma_i^2 + \sigma_\Delta^2)^{1/2}$
(d) ART = Initial RT_{NDT} + ΔRT_{NDT} + Margin (°F)
(e) All surveillance data is credible.

TABLE 9
Calculation of the ART Values for the 3/4T Location @ 16 EFPY

Material	RG 1.99 R2 Method	CF (°F)	FF	IRT_{NDT}[(a)] (°F)	ΔRT_{NDT}[(b)] (°F)	Margin[(c, e)] (°F)	ART[(d)] (°F)
Intermediate Shell Plate R-1806-1	Position 1.1	28.5	0.597	40	17	17	74
Intermediate Shell Plate R-1806-2	Position 1.1	37	0.597	0	22.1	22.1	44
Intermediate Shell Plate R-1806-3	Position 1.1	47.5	0.597	10	28.4	28.4	67
Lower Shell Plate R-1808-1	Position 1.1	37	0.597	40	22.1	22.1	84
Lower Shell Plate R-1808-2	Position 1.1	37	0.597	10	22.1	22.1	54
Lower Shell Plate R-1808-3	Position 1.1	44	0.597	40	26.3	26.3	93
	Position 2.1	42.1	0.597	40	25.1	17	82
Intermediate & Lower Shell	Position 1.1	30.7	0.597	-60	18.3	18.3	-23
Longitudinal Weld Seams (Heat # 4P6052)	Position 2.1	13.2	0.597	-60	7.9	7.9	-44
Inter. to Lower Shell Girth	Position 1.1	30.7	0.597	-60	18.3	18.3	-23
Weld Seam (Heat # 4P6052)	Position 2.1	13.2	0.597	-60	7.9	7.9	-44

Notes:
(a) Initial RT_{NDT} values are measured values except for the welds.
(b) $\Delta RT_{NDT} = CF * FF$
(c) $M = 2 * (\sigma_i^2 + \sigma_\Delta^2)^{1/2}$
(d) ART = Initial $RT_{NDT} + \Delta RT_{NDT}$ + Margin (°F)
(e) All surveillance data is credible.

TABLE 10
Calculation of the ART Values for the 1/4T Location @ 20 EFPY

Material	RG 1.99 R2 Method	CF (°F)	FF	IRT_{NDT}[(a)] (°F)	ΔRT_{NDT}[(b)] (°F)	Margin[(c, e)] (°F)	ART[(d)] (°F)
Intermediate Shell Plate R-1806-1	Position 1.1	28.5	0.934	40	26.8	26.8	94
Intermediate Shell Plate R-1806-2	Position 1.1	37	0.934	0	34.6	34	69
Intermediate Shell Plate R-1806-3	Position 1.1	47.5	0.934	10	44.4	34	88
Lower Shell Plate R-1808-1	Position 1.1	37	0.934	40	34.6	34	109
Lower Shell Plate R-1808-2	Position 1.1	37	0.934	10	34.6	34	79
Lower Shell Plate R-1808-3	Position 1.1	44	0.934	40	41.1	34	115
	Position 2.1	42.1	0.934	40	39.3	17	96
Intermediate & Lower Shell	Position 1.1	30.7	0.934	-60	28.7	28.7	-3
Longitudinal Weld Seams (Heat # 4P6052)	Position 2.1	13.2	0.934	-60	12.3	12.3	-35
Inter. to Lower Shell Girth	Position 1.1	30.7	0.934	-60	28.7	28.7	-3
Weld Seam (Heat # 4P6052)	Position 2.1	13.2	0.934	-60	12.3	12.3	-35

Notes:
(a) Initial RT_{NDT} values are measured values except for the welds.
(b) $\Delta RT_{NDT} = CF * FF$
(c) $M = 2 * (\sigma_i^2 + \sigma_\Delta^2)^{1/2}$
(d) ART = Initial RT_{NDT} + ΔRT_{NDT} + Margin (°F)
(e) All surveillance data is credible.

TABLE 11
Calculation of the ART Values for the 3/4T Location @ 20 EFPY

Material	RG 1.99 R2 Method	CF (°F)	FF	IRT_{NDT}[(a)] (°F)	ΔRT_{NDT}[(b)] (°F)	Margin[(c, e)] (°F)	ART[(d)] (°F)
Intermediate Shell Plate R-1806-1	Position 1.1	28.5	0.653	40	18.6	18.6	77
Intermediate Shell Plate R-1806-2	Position 1.1	37	0.653	0	24.2	24.2	48
Intermediate Shell Plate R-1806-3	Position 1.1	47.5	0.653	10	31.0	31.0	72
Lower Shell Plate R-1808-1	Position 1.1	37	0.653	40	24.2	24.2	88
Lower Shell Plate R-1808-2	Position 1.1	37	0.653	10	24.2	24.2	58
Lower Shell Plate R-1808-3	Position 1.1	44	0.653	40	28.7	28.7	97
	Position 2.1	42.1	0.653	40	27.5	17	85
Intermediate & Lower Shell Longitudinal Weld Seams (Heat # 4P6052)	Position 1.1	30.7	0.653	-60	20.0	20.0	-20
	Position 2.1	13.2	0.653	-60	8.6	8.6	-43
Inter. to Lower Shell Girth Weld Seam (Heat # 4P6052)	Position 1.1	30.7	0.653	-60	20.0	20.0	-20
	Position 2.1	13.2	0.653	-60	8.6	8.6	-43

Notes:
(a) Initial RT_{NDT} values are measured values except for the welds.
(b) $\Delta RT_{NDT} = CF * FF$
(c) $M = 2 * (\sigma_I^2 + \sigma_\Delta^2)^{1/2}$
(d) ART = Initial RT_{NDT} + ΔRT_{NDT} + Margin (°F)
(e) All surveillance data is credible.

The lower shell plate R-1808-1 is the limiting beltline material for the 1/4T and 3/4T locations at both 16 and 20 EFPY. Contained in Table 12 is a summary of the limiting ARTs to be used in the generation of the Seabrook Unit 1 reactor vessel heatup and cooldown curves.

TABLE 12
Summary of the Limiting ART Values Used in the
Generation of the Seabrook Unit 1 Heatup/Cooldown Curves

¼ T Limiting ART	¾ T Limiting ART
16 EFPY	
104	84
20 EFPY	
109	88

5 HEATUP AND COOLDOWN PRESSURE-TEMPERATURE LIMIT CURVES

Pressure-temperature limit curves for normal heatup and cooldown of the primary reactor coolant system have been calculated for the pressure and temperature in the reactor vessel beltline region using the methods discussed in Sections 3.0 and 4.0 of this report. This approved methodology is also presented in WCAP-14040-NP-A, Revision 2 with exception to those items discussed in Section 1 of this report.

Figures 1 and 3 present the limiting heatup curves without margins for possible instrumentation errors using heatup rates of 80 and 100°F/hr applicable for the first 16 and 20 EFPY, respectively. These curves were generated using the1996 ASME Code Section XI, Appendix G with the limiting ARTs. Figures 2 and 4 present the limiting cooldown curves without margins for possible instrumentation errors using cooldown rates of 0, 20, 40, 60, 80 and 100°F/hr applicable for 16 and 20 EFPY, respectively. Again, this curve was generated using the1996 ASME Code Section XI, Appendix G with the limiting ARTs. Allowable combination of temperature and pressure for specific temperature change rates are below and to the right of the limit lines shown in Figures 1 through 4. This is in addition to other criteria which must be met before the reactor is made critical, as discussed below in the following paragraphs.

The reactor must not be made critical until pressure-temperature combinations are to the right of the criticality limit line shown in Figures 1 and 3. The straight-line portion of the criticality limit is at the minimum permissible temperature for the 2485 psig inservice hydrostatic test as required by Appendix G to 10 CFR Part 50. The governing equation for the hydrostatic test is defined in Code Case N-641[3] (approved in February 1999) as follows:

$$1.5\ K_{Im} < K_{Ic}$$

where,

K_{Im} is the stress intensity factor covered by membrane (pressure) stress,

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

T is the minimum permissible metal temperature, and

RT_{NDT} is the metal reference nil-ductility temperature.

The criticality limit curve specifies pressure-temperature limits for core operation to provide additional margin during actual power production as specified in Reference 10. The pressure-temperature limits for core operation (except for low power physics tests) are that the reactor vessel must be at a temperature equal to or higher than the minimum temperature required for the inservice hydrostatic test, and at least 40°F higher than the minimum permissible temperature in the corresponding pressure-temperature curve for heatup and cooldown calculated as described in Section 3.0 of this report. For the heatup and cooldown curves without margins for instrumentation errors, the minimum temperature for the in service hydrostatic leak tests for the Seabrook Unit 1 reactor vessel at 16 and 20 EFPY is 164°F and 169°F, respectively. The vertical line drawn from these points on the pressure-temperature curve, intersecting a curve 40°F higher than the pressure-temperature limit curve, constitutes the limit for core operation for the reactor vessel.

Figures 1 through 4 define all of the above limits for ensuring prevention of nonductile failure for the Seabrook Unit 1 reactor vessel at 16 and 20 EFPY. The data points used for the heatup and cooldown pressure-temperature limit curves shown in Figures 1 through 4 are presented in Tables 13 through 16.

MATERIAL PROPERTY BASIS

LIMITING MATERIAL: LOWER SHELL PLATE R-1808-1
LIMITING ART VALUES AT 16 EFPY: 1/4T, 104°F
3/4T, 84°F

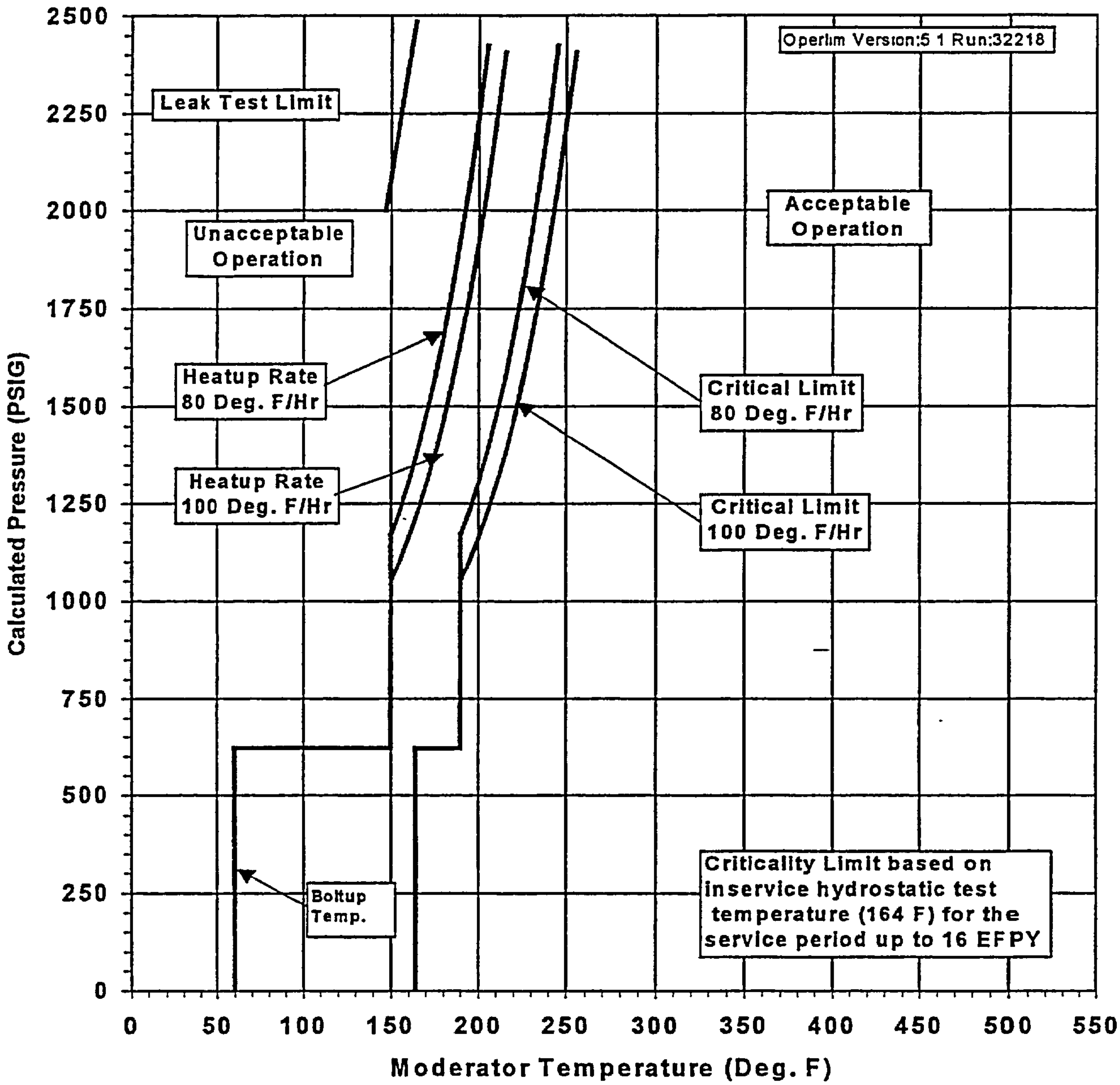

Figure 1 Seabrook Unit 1 Reactor Coolant System Heatup Limitations (Heatup Rate of 80 & 100°F/hr) Applicable for the First 16 EFPY (Without Margins for Instrumentation Errors) Using 1996 App.G Methodology w/Kic

MATERIAL PROPERTY BASIS

LIMITING MATERIAL: LOWER SHELL PLATE R-1808-1
LIMITING ART VALUES AT 16 EFPY: 1/4T, 104°F
3/4T, 84°F

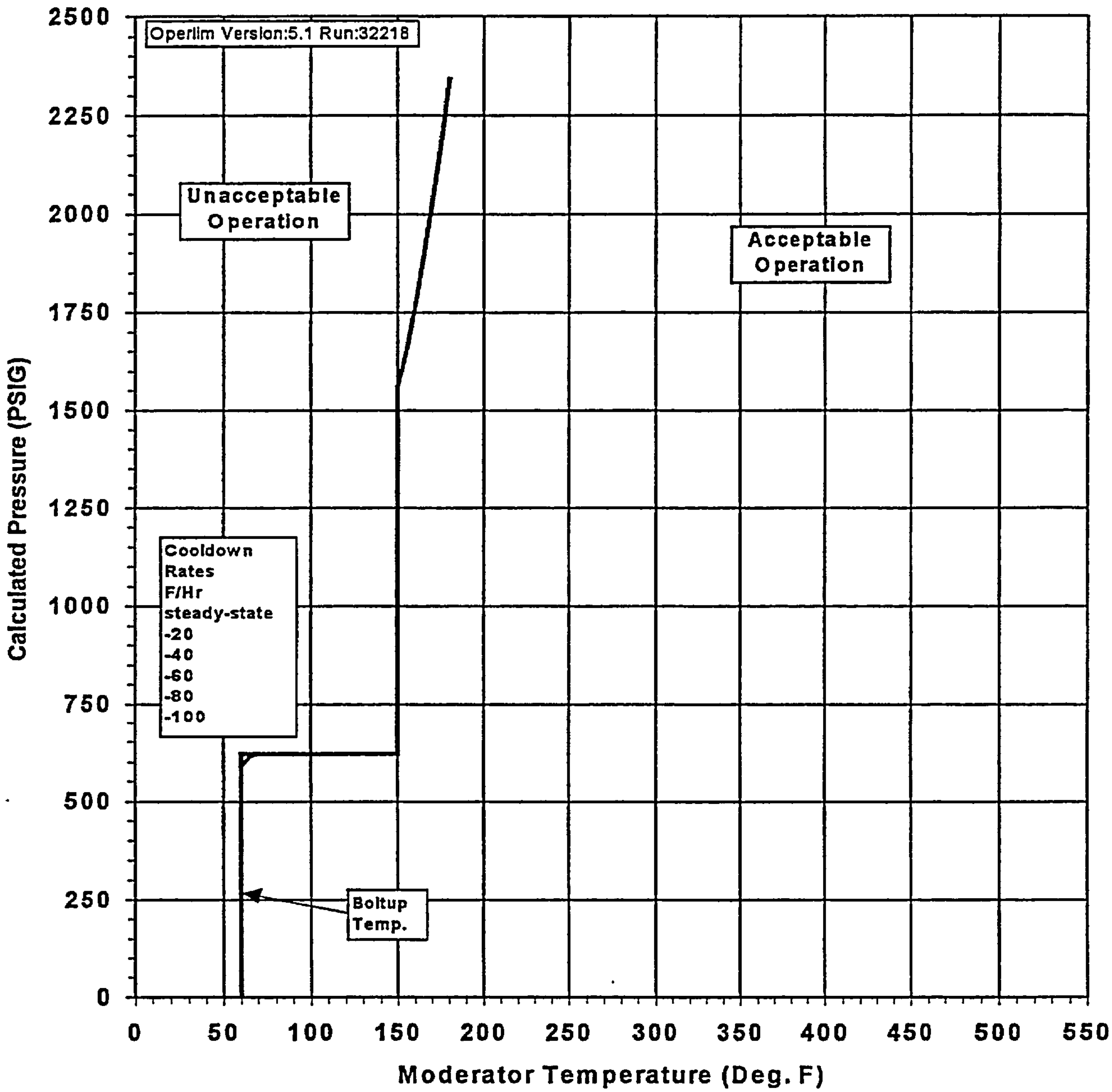

Figure 2 **Seabrook Unit 1 Reactor Coolant System Cooldown Limitations (Cooldown Rates up to 100°F/hr) Applicable for the First 16 EFPY (Without Margins for Instrumentation Errors) Using 1996 App.G Methodology w/Kic**

TABLE 13

16 EFPY Heatup Curve Data Points Using 1996 App. G w/Kic
(without Uncertainties for Instrumentation Errors)

Heatup Curves (Run # 32218)									
80 Heatup		80 Critical Limit		100 Heatup		100 Critical Limit		Leak Test Limit	
T	P	T	P	T	P	T	P	T	P
60	0	164	0	60	0	164	0	147	2000
60	621	164	620	60	621	164	620	164	2485
65	621	164	620	65	621	164	620		
70	621	164	620	70	621	164	620		
75	621	164	620	75	621	164	620		
80	621	164	620	80	621	164	620		
85	621	164	620	85	621	164	620		
90	621	164	620	90	621	164	620		
95	621	164	620	95	621	164	620		
100	621	164	620	100	621	164	620		
105	621	164	620	105	621	164	620		
110	621	164	620	110	621	164	620		
115	621	164	620	115	621	164	620		
120	621	165	620	120	621	165	620		
125	621	170	620	125	621	170	620		
130	621	175	620	130	621	175	620		
135	621	180	620	135	621	180	620		
140	621	185	620	140	621	185	620		
145	621	190	620	145	621	190	620		
150	621	190	1170	150	621	190	1053		
150	621	195	1235	150	621	195	1105		
150	1170	200	1307	150	1053	200	1162		
155	1235	205	1387	155	1105	205	1227		
160	1307	210	1475	160	1162	210	1298		
165	1387	215	1574	165	1227	215	1378		
170	1475	220	1682	170	1298	220	1466		
175	1574	225	1803	175	1378	225	1563		
180	1682	230	1935	180	1466	230	1671		
185	1803	235	2082	185	1563	235	1790		
190	1935	240	2244	190	1671	240	1922		
195	2082	245	2423	195	1790	245	2068		
200	2244			200	1922	250	2228		
205	2423			205	2068	255	2406		
				210	2228				
				215	2406				

TABLE 14

16 EFPY Cooldown Curve Data Points Using 1996 App. G w/Kic
(without Uncertainties for Instrumentation Errors)

Cooldown Curves (32218)											
Steady State		20°F/hr. Rate		40°F/hr Rate		60°F/hr. Rate		80°F/hr. Rate		100°F/hr. Rate	
T	P	T	P	T	P	T	P	T	P	T	P
60	0	60	0	60	0	60	0	60	0	60	0
60	621	60	621	60	621	60	621	60	621	60	588
65	621	65	621	65	621	65	621	65	621	65	614
70	621	70	621	70	621	70	621	70	621	70	621
75	621	75	621	75	621	75	621	75	621	75	621
80	621	80	621	80	621	80	621	80	621	80	621
85	621	85	621	85	621	85	621	85	621	85	621
90	621	90	621	90	621	90	621	90	621	90	621
95	621	95	621	95	621	95	621	95	621	95	621
100	621	100	621	100	621	100	621	100	621	100	621
105	621	105	621	105	621	105	621	105	621	105	621
110	621	110	621	110	621	110	621	110	621	110	621
115	621	115	621	115	621	115	621	115	621	115	621
120	621	120	621	120	621	120	621	120	621	120	621
125	621	125	621	125	621	125	621	125	621	125	621
130	621	130	621	130	621	130	621	130	621	130	621
135	621	135	621	135	621	135	621	135	621	135	621
140	621	140	621	140	621	140	621	140	621	140	621
145	621	145	621	145	621	145	621	145	621	145	621
150	621	150	621	150	621	150	621	150	621	150	621
150	621	150	621	150	621	150	621	150	621	150	621
150	1560	150	1560	150	1560	150	1560	150	1560	150	1560
155	1660	155	1660	155	1660	155	1660	155	1660	155	1660
160	1771	160	1771	160	1771	160	1771	160	1771	160	1771
165	1893	165	1893	165	1893	165	1893	165	1893	165	1893
170	2028	170	2028	170	2028	170	2028	170	2028	170	2028
175	2178	175	2178	175	2178	175	2178	175	2178	175	2178
180	2343	180	2343	180	2343	180	2343	180	2343	180	2343

MATERIAL PROPERTY BASIS

LIMITING MATERIAL: LOWER SHELL PLATE R-1808-1
LIMITING ART VALUES AT 20 EFPY: 1/4T, 109°F
3/4T, 88°F

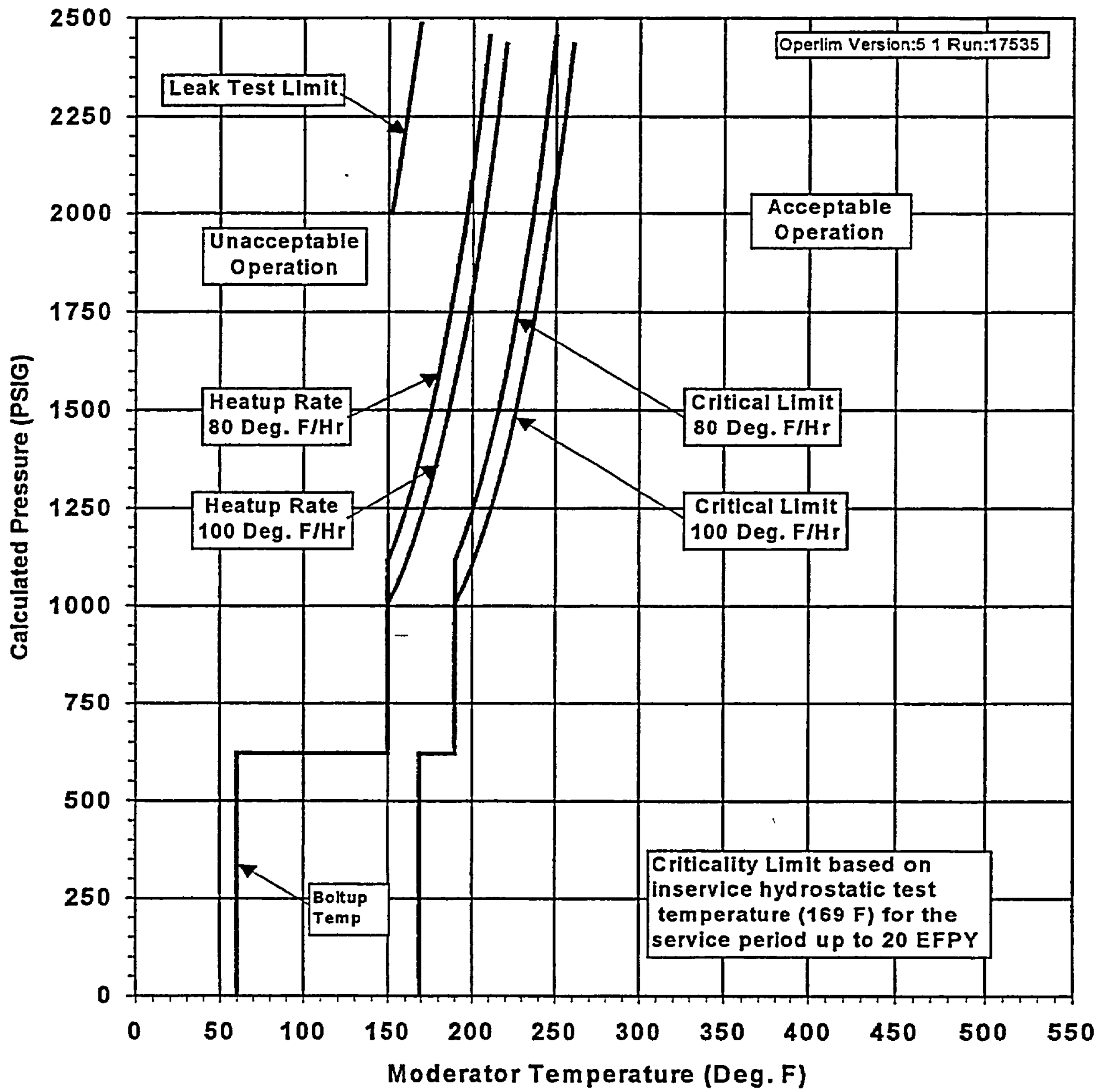

Figure 3 Seabrook Unit 1 Reactor Coolant System Heatup Limitations (Heatup Rate of 80 & 100°F/hr) Applicable for the First 20 EFPY (Without Margins for Instrumentation Errors) Using 1996 App.G Methodology w/Kic

MATERIAL PROPERTY BASIS

LIMITING MATERIAL: LOWER SHELL PLATE R-1808-1
LIMITING ART VALUES AT 20 EFPY: 1/4T, 109°F
3/4T, 88°F

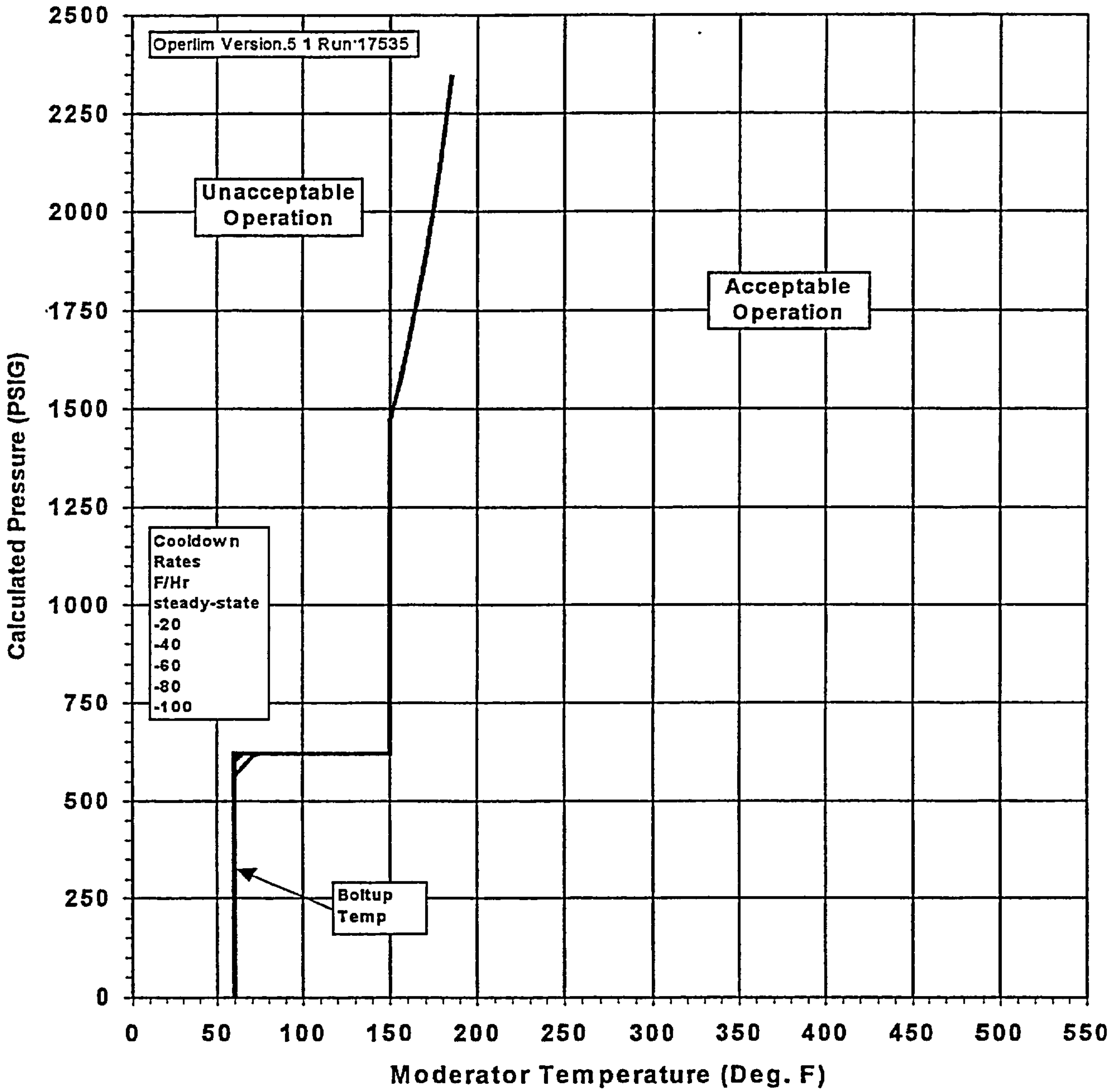

Figure 4 Seabrook Unit 1 Reactor Coolant System Cooldown Limitations (Cooldown Rates up to 100°F/hr) Applicable for the First 20 EFPY (Without Margins for Instrumentation Errors) Using 1996 App.G Methodology w/Kic

TABLE 15

20 EFPY Heatup Curve Data Points Using 1996 App. G w/Kic
(without Uncertainties for Instrumentation Errors)

Heatup Curves (Run # 17535)									
80 Heatup		**80 Critical Limit**		**100 Heatup**		**100 Critical Limit**		**Leak Test Limit**	
T	**P**	**T**	**P**	**T**	**P**	**T**	**P**	**T**	**P**
60	0	169	0	60	0	169	0	152	2000
60	621	169	620	60	621	169	620	169	2485
65	621	169	620	65	621	169	620		
70	621	169	620	70	621	169	620		
75	621	169	620	75	621	169	620		
80	621	169	620	80	621	169	620		
85	621	169	620	85	621	169	620		
90	621	169	620	90	621	169	620		
95	621	169	620	95	621	169	620		
100	621	169	620	100	621	169	620		
105	621	169	620	105	621	169	620		
110	621	169	620	110	621	169	620		
115	621	169	620	115	621	169	620		
120	621	169	620	120	621	169	620		
125	621	170	620	125	621	170	620		
130	621	175	620	130	621	175	620		
135	621	180	620	135	621	180	620		
140	621	185	620	140	621	185	620		
145	621	190	620	145	621	190	620		
150	621	190	1116	150	621	190	1006		
150	621	195	1176	150	621	195	1054		
150	1116	200	1242	150	1006	200	1107		
155	1176	205	1316	155	1054	205	1166		
160	1242	210	1398	160	1107	210	1232		
165	1316	215	1488	165	1166	215	1305		
170	1398	220	1588	170	1232	220	1386		
175	1488	225	1699	175	1305	225	1476		
180	1588	230	1822	180	1386	230	1575		
185	1699	235	1957	185	1476	235	1685		
190	1822	240	2107	190	1575	240	1807		
195	1957	245	2272	195	1685	245	1941		
200	2107	250	2454	200	1807	250	2089		
205	2272			205	1941	255	2253		
210	2454			210	2089	260	2433		
				215	2253				
				220	2433				

TABLE 16

20 EFPY Cooldown Curve Data Points Using 1996 App. G w/Kic
(without Uncertainties for Instrumentation Errors)

Cooldown Curves (17535)											
Steady State		20°F/hr. Rate		40°F/hr Rate		60°F/hr. Rate		80°F/hr. Rate		100°F/hr. Rate	
T	P	T	P	T	P	T	P	T	P	T	P
60	0	60	0	60	0	60	0	60	0	60	0
60	621	60	621	60	621	60	621	60	620	60	564
65	621	65	621	65	621	65	621	65	621	65	587
70	621	70	621	70	621	70	621	70	621	70	614
75	621	75	621	75	621	75	621	75	621	75	621
80	621	80	621	80	621	80	621	80	621	80	621
85	621	85	621	85	621	85	621	85	621	85	621
90	621	90	621	90	621	90	621	90	621	90	621
95	621	95	621	95	621	95	621	95	621	95	621
100	621	100	621	100	621	100	621	100	621	100	621
105	621	105	621	105	621	105	621	105	621	105	621
110	621	110	621	110	621	110	621	110	621	110	621
115	621	115	621	115	621	115	621	115	621	115	621
120	621	120	621	120	621	120	621	120	621	120	621
125	621	125	621	125	621	125	621	125	621	125	621
130	621	130	621	130	621	130	621	130	621	130	621
135	621	135	621	135	621	135	621	135	621	135	621
140	621	140	621	140	621	140	621	140	621	140	621
145	621	145	621	145	621	145	621	145	621	145	621
150	621	150	621	150	621	150	621	150	621	150	621
150	621	150	621	150	621	150	621	150	621	150	621
150	1469	150	1469	150	1469	150	1469	150	1469	150	1469
155	1560	155	1560	155	1560	155	1560	155	1560	155	1560
160	1660	160	1660	160	1660	160	1660	160	1660	160	1660
165	1771	165	1771	165	1771	165	1771	165	1771	165	1771
170	1893	170	1893	170	1893	170	1893	170	1893	170	1893
175	2028	175	2028	175	2028	175	2028	175	2028	175	2028
180	2178	180	2178	180	2178	180	2178	180	2178	180	2178
185	2343	185	2343	185	2343	185	2343	185	2343	185	2343

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ENCLOSURE 3 TO NYN-02093

NFSB 02-0061

Seabrook Station
Cold Overpressure Mitigating System (COMS)
Setpoint Development Methodology

P. J. Guimond

August 2002

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SEABROOK STATION
COLD OVERPRESSURE MITIGATING SYSTEM (COMS) SETPOINTS

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DISCLAIMER OF RESPONSIBILITY

ABSTRACT

This report documents the considerations and selection procedures used to determine setpoints for the Cold Overpressure Mitigating System (COMS) at Seabrook Station. COMS limits reactor vessel pressure during cold overpressurization events to values below the low temperature overpressurization (LTOP) limits. The LTOP limits are in part based on the reactor vessel heatup and cooldown pressure/temperature limits specified by 10 CFR 50 Appendix G. The Appendix G limits shift as reactor vessel neutron irradiation increases over time. Appendix G limits and COMS setpoints are derived to conservatively bound the limits through a specified vessel total exposure. Prior to exceeding the applicable exposure, the Appendix G limits and COMS setpoints must be modified to conservatively bound a higher total exposure. This report describes the application of procedures and associated design-basis event evaluations to support modifications to the COMS setpoints for Seabrook Station.

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1.0 INTRODUCTION AND SUMMARY

1.1 Introduction

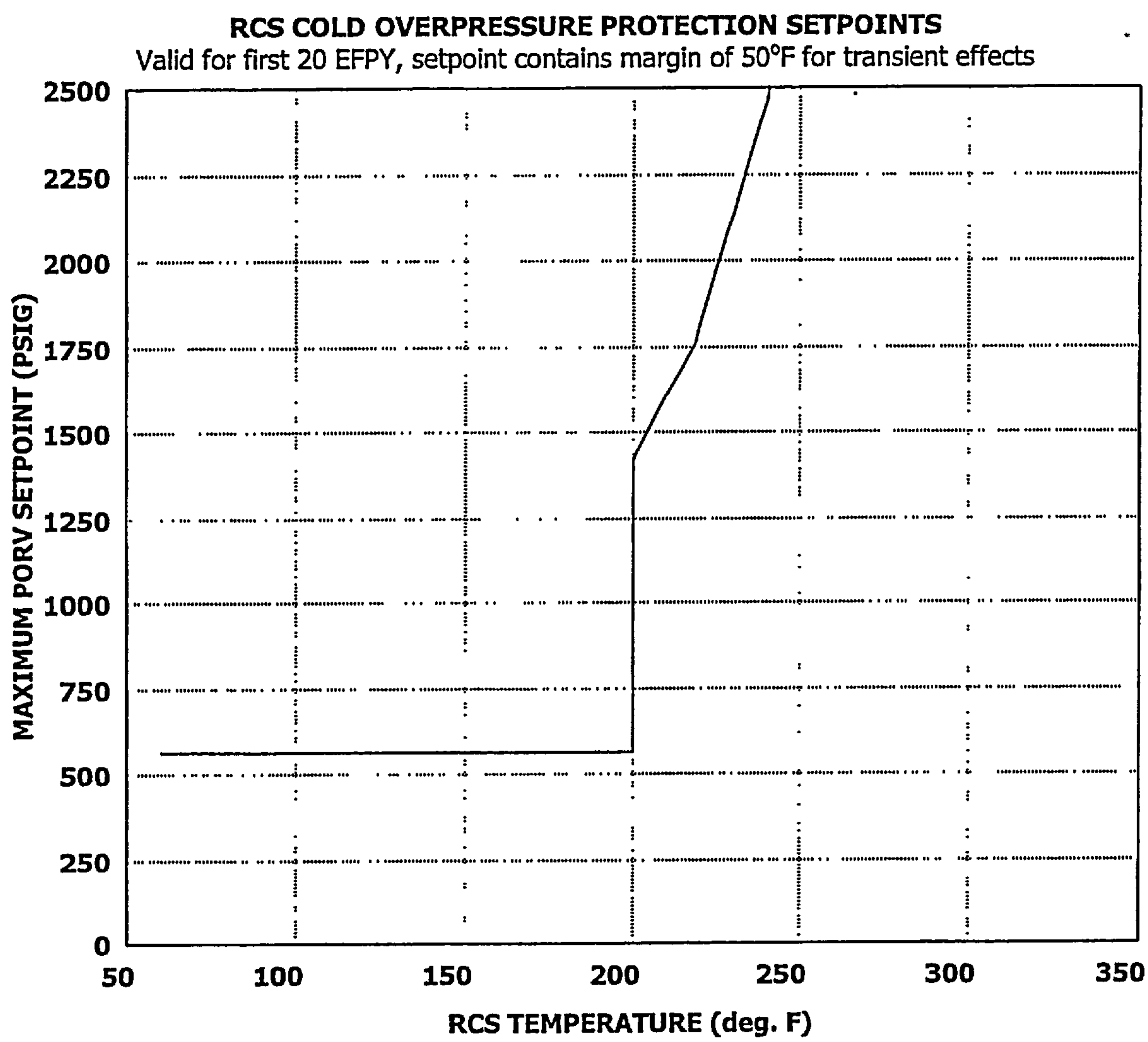
One alternative for providing low temperature overpressure protection allowed by Seabrook's Technical Specifications is the pressurizer pressure operated relief valves (PORVs) operating with reduced pressure setpoints. The reduced pressure setpoints are a function of the indicated reactor coolant system temperature. The PORVs and reduced pressure setpoint functions are referred to as the Cold Overpressure Mitigating System (COMS). This report documents the procedure used to develop the temperature dependent pressure setpoints for the COMS at Seabrook Station. The topics herein addressed are:

- Pressure limit selection
- Design basis event evaluations
- COMS setpoint selection
- Application of setpoint uncertainties
- COMS arming temperature

1.2 Summary

The reactor vessel heatup and cooldown limitations for Seabrook have been updated to cover an increased total exposure of 20 EFPY [1]. Revised pressurizer PORV COMS setpoints have therefore been derived for operation of Seabrook Station's reactor vessel to a cumulative exposure of 20 EFPY (see Figure 1-1). The considerations included in the derivation of these setpoints are described in section 2.0 of this report. The revised setpoints provide overpressure protection for the Seabrook reactor vessel and closure head/flange consistent with the requirements of ASME Section XI Division I Appendix G [6], ASME Code Case N-641 [2], and 10 CFR Part 50 Appendix G [3]. Implementation of these revised maximum allowable COMS setpoints requires USNRC approval of an exemption from 10 CFR Part 50 Appendix G LTOP pressure limit requirements for application of ASME Code case N-641.

Figure 1-1



$T \leq 200.0$, $P = 561.0$; ($200.0 < T \leq 218.65$), $P = 18.2*T - 2221.0$; $T > 218.65$, $P = 33.0*T - 5457.0$

2.0 SETPOINT SELECTION CONSIDERATIONS

2.1 Pressure Limit Selection

New Seabrook Unit 1 heat up and cool down allowable pressure-temperature limit curves for normal operation [1] were derived using the K_{Ic} methodology allowed by ASME Code Case N-641[2]. Per Code Case N-641, to provide protection against failure during reactor startup and shutdown operation due to overpressure events that have been classified Service Level A or B, low temperature overpessure protection (LTOP) systems shall limit the maximum pressure in the vessel to 100% of the pressure determined to satisfy the equation, $2K_{Im} + K_{It} < K_{Ic}$. Credit is taken for the fact that experience shows that LTOP events are most likely to occur at isothermal conditions. The allowable pressure limit for the COMS is therefore taken to be the steady-state (isothermal) pressure limit curve (see Table 2-1). Below 150°F, to comply with the closure head/vessel flange region limitation on system pressure imposed by 10 CFR Part 50, Appendix G [3] the allowable pressure is limited to a maximum of 621 psig.

2.2 Design Basis Event Evaluations

Two design basis overpressure events are considered in the COMS setpoint development: a heatup transient and a mass addition event.

The COMS design basis heatup transient starts from an initial condition where the RCS flow rate is at zero, the steam generators are at a temperature 50°F hotter than the rest of the RCS (limited by Technical Specification 3.4.1.4.1), the pressurizer is water solid (a bubble has not yet been formed) and then a RCP is inadvertently started. The starting of the RCP initiates heat transfer from the secondary to the primary side which causes a pressurization of the primary side. The pressure response to the heatup transient is analyzed using a modified version of the PRESS computer code [4]. The modifications to the code were made to provide increased flexibility in the modeling of relief valve opening/closing dynamics (delay time, non-linear opening/closing characteristics), control input location (static and dynamic spatial pressure differences), and type of relief valve (PORV vs RHR safety valve). The basic pressure and heat transfer solution techniques were not modified.

SEABROOK STATION
COLD OVERPRESSURE MITIGATING SYSTEM (COMS) SETPOINTS

TABLE 2-1

LTOP Pressure Limit for 20 EFPY

RCS Temperature (°F)	Steady-State Pressure Limit (psig)	RV Flange Limit (psig)	LTOP Pressure Limit (psig)
60	0	0	**0**
60	564[1]	621	**621**
65	587[1]	621	**621**
70	614[1]	621	**621**
75	621	621	**621**
80	621	621	**621**
85	621	621	**621**
90	621	621	**621**
95	621	621	**621**
100	621	621	**621**
105	621	621	**621**
110	621	621	**621**
115	621	621	**621**
120	621	621	**621**
125	621	621	**621**
130	621	621	**621**
135	621	621	**621**
140	621	621	**621**
145	621	621	**621**
150	621	-	**621**
150	1469	-	**1469**
155	1560	-	**1560**
160	1660	-	**1660**
165	1771	-	**1771**
170	1893	-	**1893**
175	2028	-	**2028**
180	2178	-	**2178**
185	2343	-	**2343**
190	2508 (extrapolated)[1]	-	**2508**
195	2673 (extrapolated)[1]	-	**2673**

[1] Since the slope of limit increases with temperature, linear extrapolation is conservative.

The PRESS computer code models the reactor coolant system, including the pressurizer, the primary and secondary sides of the steam generator tube bundle region, and the steam generator tube walls. The Seabrook PRESS model nodalization is shown in Figure 2-1.

To determine the system pressure transient following the initiation of flow from the reactor coolant pump, PRESS dynamically calculates the primary/secondary heat transfer through the steam generator tube walls, the fluid energy transfer through the primary system via coolant flow, and the discharge rate through the relief valve. For the Seabrook analysis, the loop flows were assumed to increase linearly from zero to steady-state conditions in 15 seconds.

In addition to simulating the primary/secondary heat transfer, the PRESS model simulates the dynamic opening and closing characteristics of the PORV relief valves. The spatial and dynamic pressure drops affecting the wide range pressure input to the COMS PORV control system and the PORV inlet pressure are also accounted for in the modified PRESS model.

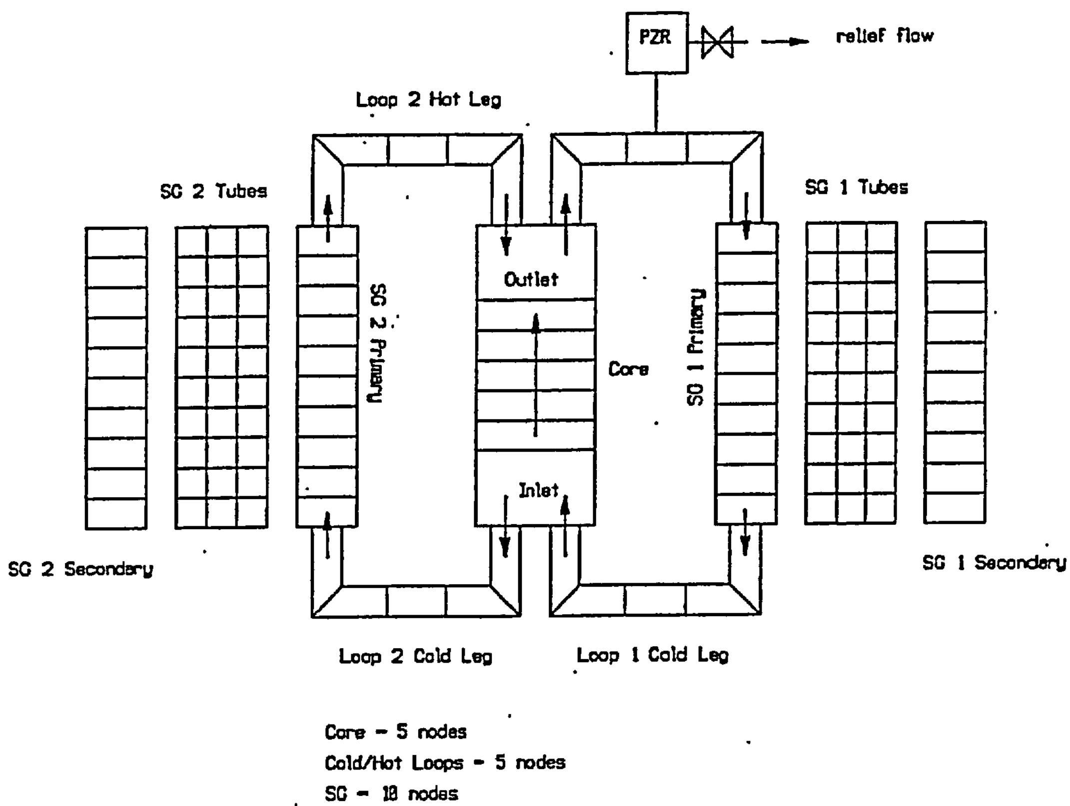
The PORV relief flow as a function of valve inlet pressure was obtained from an independent calculation of PORV performance for a range of subcooled and saturated liquid relief conditions using the RETRAN-02 computer code [5]. The RETRAN-02 analysis used the iso-enthalpic expansion (IHE) flow model to conservatively minimize the predicted relief flow.

PRESS simulations of the heatup event were used to determine the peak RCS pressure setpoint overshoot defined as the peak pressure in the RV downcomer adjacent to the beltline minus the nominal PORV setpoint. Calculations were performed over a range of assumed initial RCS water temperatures below the expected COMS arming temperature.

The second design basis event is the mass addition event. The event considered is mass addition from all potentially operating charging pumps via both the normal charging flow path, with the charging flow (FCV-121) and head (HCV-182) control valves fully open, and the charging pump safety injection flow path in parallel, due to an inadvertent opening of one of the two SI flow path block valves. Letdown flow was assumed to be isolated. Current plant Technical Specifications limit the number of operable charging pumps to a single CCP in Modes 4 and 5 except for a brief transition period when entering or exiting Mode 3 and/or when swapping pumps. Therefore the analysis assumed flow from a single CCP.

FIGURE 2-1

SEABROOK PRESS MODEL NODALIZATION



Loop 1 models the single loop where RCP is started, Loop2 models the three loops with inactive RCPs.

The mass addition event was analyzed using a RETRAN-02 model of the Seabrook reactor coolant system to determine the pressure response and maximum RCS pressure setpoint overshoot. Maximum setpoint overshoot was determined as a function of initial wide-range pressure and PORV setpoint for a cold (100°F) reactor coolant condition. Assuming a cold RCS conservatively minimizes the compressibility of the RCS water volume, maximizing the rate of pressure increase and setpoint pressure overshoot.

2.3 COMS Setpoint Determination

The pressure overshoot values calculated for the design basis heat and mass addition events are used to determine the maximum allowable PORV setpoint required to prevent violation of the LTOP allowable pressure limit during the event. The maximum PORV setpoint must be a value lower than the selected LTOP allowable pressure limit minus the overshoot predicted for the applicable temperature condition (determination of the applicable temperature for each event is discussed below).

In the Seabrook COMS design the auctioneered low wide-range hot leg temperature is used as input to the setpoint function generator for one PORV and the auctioneered low wide-range cold leg temperature is used as input to the setpoint function generator for the second PORV.

During the design basis heatup event, the temperature in the cold legs may increase by as much as 50°F due to heat addition to the loop flow from the hotter steam generator secondary water once the RCP is started. This heated water eventually reaches all four hot legs and thus the setpoint inputs for auctioneered low cold and hot leg temperature are expected to increase during the event. The temperature increase is limited to the temperature of the steam generator secondary water, so the maximum temperature increase is limited to 50°F. Since the PORV setpoints are a programmed function of the input temperature, the PORV setpoint may increase during the event due to the increase in indicated RCS temperatures. To account for this possibility, the COMS setpoints are chosen such that the maximum PORV setpoint at a temperature up to 50°F higher than the initial RCS temperature is low enough to preclude violation of the LTOP pressure limit considering the maximum pressure overshoot for the event. The setpoint at the initial RCS temperature plus 50°F must be less than or equal to the LTOP pressure limit at the initial temperature minus the maximum pressure overshoot.

During a mass addition event, the injection of cold water downstream of the location in the cold leg where RCS wide-range temperature is measured could result in lower fluid temperatures adjacent to the RV wall in the downcomer region than the temperature indicated by the cold leg temperature setpoint input. Analysis of the mixed stream fluid temperature entering the reactor vessel showed that the maximum temperature difference is less than 50°F for the maximum injection flow with RHR flow rate at a conservatively low minimum value. Therefore, to prevent pressure from exceeding the LTOP allowable pressure limit applicable to the potentially lower temperature at the vessel beltline, the maximum COMS PORV setpoint at any indicated wide-range temperature must be less than or equal to the LTOP pressure limit corresponding to a temperature 50°F lower, minus the maximum pressure overshoot for a mass addition event at any temperature over the 50°F temperature span. This is the same condition determined to be required for the heat addition event.

The heat and mass addition event setpoint overshoot values are also used to indicate the amount of stagger that should be maintained between the two PORV setpoints. The COMS system consists of two independently actuated PORVs. In order to minimize the potential undershoot of the setpoint (which could cause RCS pressure to challenge minimum RCP seal operating pressure limits) it is desirable for only a single valve to be actuated as the result of an overpressure event. Since a single valve has sufficient relief capacity to limit the RCS pressure to the setpoint plus overshoot, setting one valve to a setpoint less than the setpoint of the other valve minus the overshoot ideally eliminates the challenge to the other valve.

A second factor influencing the selection of the lower PORV setpoint is the effect of pressurizer water temperature on PORV relief volumetric relief rate. The PORV volumetric relief rate was found to decrease as the enthalpy of the pressurizer liquid approached the saturation temperature corresponding to the PORV setpoint. As a result, the pressure overshoot for a mass addition event was found to increase substantially when the temperature of the fluid in the pressurizer exceeds the value which would result in the PORV volumetric relief capacity at the setpoint pressure dropping below the mass addition rate from the CCP at the corresponding RCS pressure. Therefore, in order to minimize the overshoot, the maximum initial pressurizer liquid enthalpy/temperature is limited to a value which assures that the fluid relieved at the PORV setpoint is adequately subcooled, resulting in a relief valve flow at the maximum allowed setpoint pressure greater than or equal to the mass addition rate at that pressure. The mechanism used to implement this limit is to set or assure that the nominal COMS setpoint of the PORV with the lower setpoint limits pressurizer water temperature to the required maximum value. The setpoint of the PORV with the minimum COMS setpoint is indicated in station operating procedures for heatup and cooldown. This assures that the pressurizer water is adequately subcooled when the maximum setpoint pressure is reached, minimizing the pressure overshoot.

Within these constraints, the PORV setpoints are set as high as possible in order to maximize the margin available between the PORV setpoint and the RCP seal minimum operating pressure limit (325 psig). Maximizing this margin minimizes the chance of damaging a seal due to pressure undershoot while the relief valve is closing. Operation of the RCP at RCS pressures lower than the limit could result in damage to the seal. However, preventing damage to the seal is an economic concern and not a safety limit. Preventing non-ductile failure of the vessel is the primary safety concern. Therefore, if necessary to prevent violating the LTOP allowable pressure limit, the COMS setpoint is set below the value which would challenge the RCP seal limit, since the seal limit is of secondary significance.

Table 2-2 provides an example of the determination of the maximum COMS PORV setpoint for the new LTOP pressure limit for 20 EFPY for Seabrook. The maximum PORV setpoint, P, specified in TS Figure 3.4-4 as a function of RCS temperature, T, is expressed as a set of linear equations defining a lower bounding fit of the last two columns in Table 2-2:

$$
\begin{array}{ll}
T \leq 200.0^{\circ}F, & P = 561.0 \text{ psig;} \\
(200.0^{\circ}F < T \leq 218.65^{\circ}F), & P = 18.2{*}T - 2221.0 \text{ psig;} \\
T > 218.65^{\circ}F, & P = 33.0{*}T - 5457.0 \text{ psig}
\end{array}
$$

The equations conservatively fit the allowable setpoint limit as shown in Table 2-3.

SEABROOK STATION
COLD OVERPRESSURE MITIGATING SYSTEM (COMS) SETPOINTS

TABLE 2-2
Derivation of Maximum Allowable PORV COMS Setpoint for 20 EFPY

RCS Temperature (°F)	LTOP Allowable Pressure Limit (psig)	Heat Addition Event Bounding Pressure Over-shoot (psi)	Mass Addition Event Bounding Pressure Over-shoot (psi)	Maximum Allowable PORV COMS Setpoint (psig)	Minimum Wide Range RCS Temperature For Setpoint (°F)
60 (min. bolt-up)	621	40	60	561	60
60	621	40	60	561	110
65	621	40	60	561	115
70	621	40	60	561	120
75	621	40	60	561	125
80	621	40	60	561	130
85	621	40	60	561	135
90	621	40	60	561	140
95	621	40	60	561	145
100	621	40	60	561	150
105	621	40	60	561	155
110	621	40	60	561	160
115	621	40	60	561	165
120	621	40	60	561	170
125	621	40	60	561	175
130	621	40	60	561	180
135	621	40	60	561	185
140	621	40	60	561	190
145	621	40	60	561	195
150	621	40	60	561	200
150	1469	40	50	1419	200
155	1560	40	50	1510	205
160	1660	40	50	1610	210
165	1771	40	50	1721	215
170	1893	40	45	1848	220
175	2028	40	45	1983	225
180	2178	40	45	2133	230
185	2343	40	45	2298	235
190	2508	40	45	2463	240
195	2673	40	45	2628	245

TABLE 2-3

Comparison of Maximum PORV Setpoint Equations to Maximum Allowable Setpoint

RCS Temperature (°F)	Maximum Allowable Setpoint (psig)	Maximum PORV Setpoint Equation (psig)
60 - 200	561	561
200	1419	1419
205	1510	1510
210	1610	1601
215	1721	1692
220	1848	1803
225	1983	1968
230	2133	2133
235	2298	2298
240	2463	2463
245	2628	2628

2.4 Application of Setpoint Uncertainties

The wide range temperature and pressure inputs to the COMS PORV setpoint function generators and the function generator and trip bistable hardware are subject to various uncertainty factors which influence setpoint accuracy, including sensor/instrument accuracy, calibration tolerance, drift, and environmental conditions. The uncertainties on the PORV setpoint are conservatively quantified and statistically combined in the determination of nominal "in-plant" setpoint equations for each PORV. The equations for the lower set PORV are also adjusted downward by the maximum overshoot to provide the desired stagger to minimize the potential for both valves opening in response to an event.

2.5 COMS Arming Temperature

The COMS system design has an arming bi-stable for each PORV. When wide-range RCS temperature goes below the selected arming temperature, the bi-stable arms COMS by causing the associated PORV block valve to open (if it is closed) and enabling the PORV COMS setpoint.

ASME Code Case N-641 indicates the LTOP System Effective Temperature is the "temperature at or above which the safety relief valves provide adequate protection against nonductile failure". Per the code case, LTOP systems shall be effective below the higher of an inlet coolant temperature of 200°F or a coolant temperature corresponding to a reactor vessel 1/4T metal temperature of RT_{NDT} + 40°F for inside axial surface flaws and RT_{NDT} – 85°F for inside circumferential surface flaws for all vessel beltline materials. The adjusted 1/4T RT_{NDT} for the limiting beltline material at 20 EFPY is 109°F. Therefore, the code case requires the COMS system to be effective below an inlet coolant temperature of 200°F.

Examination of Table 2-2 indicates that the maximum COMS PORV setpoint at 200°F is lower than the safety relief valve setpoint (2485 psig ±3%, per TS 3.4.2.1). Therefore, the COMS arming temperature is selected to be greater than or equal to the temperature at which the maximum COMS setpoint pressure limit is ≥ safety relief valve maximum opening setpoint plus an allowance for the difference in pressure at the safety valve and the indicated wide range pressure which actuates the PORV in the COMS mode. This temperature is then adjusted to a higher value to include allowances for applicable COMS setpoint uncertainties.

3.0 REFERENCES

[1] Laubham, T. J., WCAP-15745, Seabrook Unit 1 Heatup and Cooldown Limit Curves for Normal Operation, Westinghouse Non-Proprietary Class 3, December 2001.

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

[3] Code of Federal Regulations, 10 CFR Part 50, Appendix G, "Fracture Toughness Requirements," U. S. Nuclear Regulatory Commission, Washington, D. C., Federal Register, Volume 60, No. 243, dated December 19, 1995.

[4] Chapman, J. R., YAEC-1124, PRESS An Analytical Model Used in PWR Overpressurization Analysis, February 1977.

[5] EPRI NP-1850-CCM-A, Computer Code Manual, RETRAN-02 – A Program for Transient Thermal-Hydraulic Analysis of Complex Fluid Flow Systems, June 1987.