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OPPD
Omaha Public Power District

444 South 16th Street Mall
Omaha NE 68102-2247

December 14, 2001
LIC-01-0114

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

References: See Attachment 5

SUBJECT: Fort Calhoun Station Unit No. 1 License Amendment Request, "Pressure and Temperature (P-T) Limit Curve for 40 Effective Full Power Years (EFPY)"

Pursuant to 10 CFR 50.90, Omaha Public Power District (OPPD) hereby requests the following amendment to Technical Specifications (TS) 2.1.1, 2.1.2, and 2.1.6. The proposed amendment deletes TS Figures 2-1A (Reactor Coolant System (RCS) Pressure - Temperature Limits for Heatup) and 2-1B (RCS Pressure - Temperature Limits for Cooldown). Additionally, OPPD proposes to change the lowest service temperature from 182°F to 164°F to be in compliance with Reference 4, ASME Section III, NB-2332 and the basis for the minimum boltup temperature to be in compliance with Reference 5, Section XI, Appendix G. The Basis section for Technical Specification 2.1.2 is being updated to reflect these changes, the use of ASME Code Case N-640, and Westinghouse Electric Company/Combustion Engineering's P-T limit curve methodology as applicable. Finally, based on the replacement of TS Figures 2-1A and 2-1B with the single TS Figure 2-1, the following TS are required to be changed: 2.1.1(8), 2.1.2, 2.1.2(1), 2.1.2(2), 2.1.2(6), 2.1.2(6)(a), 2.1.2(6)(c), 2.1.2(6)(d), and 2.1.6(4).

OPPD's relief request to apply ASME Code Case N-640 (Reference 6) is being submitted separately for NRC approval in parallel with this amendment request.

Attachment 1 provides the No Significant Hazards Evaluation and the technical bases for these requested changes to the Technical Specifications. Attachments 2 and 3 contain a marked-up and clean version reflecting the requested Technical Specification and Basis changes. Attachment 4 contains the non-proprietary version of the references for Attachment 1.

The precedence for the change in TS Figure 2-1 is the use of ASME Code Case N-640, which the NRC has previously reviewed and approved for other utilities (e.g., Calvert Cliffs, Reference 7). Furthermore, W/CE's P-T limit curve methodology, which was used to develop TS Figure 2-1 was also previously reviewed and approved by the NRC for Indian Point 3 and for the Combustion Engineering Owners Group Topical RCS pressure and temperature limits report

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(PTLR), (References 8 and 9.) Finally, the basis for reducing the lowest service temperature and shifting the technical basis for the minimum boltup temperature is endorsed via 10 CFR 50 Appendix G and 10 CFR 50.55a the ASME Boiler and Pressure Vessel Code, 1995 Edition and addenda through the 1996 Addenda Section III and Section XI, respectively.

In order to expedite replacement of one batch of fuel prone to grid to rod fretting fuel failures with fuel assemblies not prone to the failure mechanism, OPPD requests approval of the proposed amendment by March 15, 2002, to be available for the plant cooldown at the beginning of the spring 2002 refueling outage. The spring 2002 refueling outage is planned to start on May 4, 2002. Once approved, the amendment shall be implemented by May 3, 2002.

I declare under penalty of perjury that the foregoing is true and correct. (Executed on December 14, 2001)

If you have any questions or require additional information, please contact Dr. R. L. Jaworski of my staff at 402-533-6833.

Sincerely,

W. G. Gates

W. G. Gates
Vice President

WGG/RLJ/fjj

Attachments
1. Fort Calhoun Station's Evaluation
2. Mark-up of Technical Specifications
3. Clean Version of Technical Specifications
4. References for Attachment 1(Non-proprietary Version)
   a. LTR-PS-01-26, Revision 0, "Evaluation of the Current LTOP Analysis for Revised Technical Specification P-T Limits at the Fort Calhoun Station"
   b. Report DAR-PS-01-4, Revision 0, "Reactor Coolant System Pressure-Temperature Limits and Low Temperature Overpressure Enable Temperature for 40 Effective Full Power Years for Fort Calhoun Station Unit 1"
5. References for Cover Letter

c: E. W. Merschoff, NRC Regional Administrator, Region IV
A. B. Wang, NRC Project Manager
W. C. Walker, NRC Senior Resident Inspector
Division Administrator, Public Health Assurance, State of Nebraska
Winston & Strawn

Fort Calhoun Station's Evaluation
For
Request for Pressure and Temperature Limit Curve for 40 EFPY

1.0 INTRODUCTION

This letter is a request to amend Operating License DPR-40 for the Fort Calhoun Station (FCS) Unit No. 1.

The Omaha Public Power District (OPPD) proposes to change: 1) the pressure and temperature (P-T) limit curve (Technical Specifications (TS) Figure 2-1); 2) the application of the predicted fluence and its validity to 40 effective full power years (EFPY) for TS Figure 2-1; 3) the lowest service temperature; and 4) the basis for the minimum boltup temperature from NDTT to RT_{NDT}. These revisions will allow the use of higher cooldown rates. Additionally, more operating margin will be gained, and the use of a lower minimum boltup temperature will be available when the low temperature overpressure protection (LTOP) re-analysis is completed (currently in progress) and submitted by OPPD to the NRC for review. Presently an LTOP analysis is in preparation for submittal before August 2002. NRC approval of this amendment request by March 15, 2002, with the parallel NRC approval of the relief request for ASME Code Case N-640, and implementation of this amendment by May 3, 2002, is desired by OPPD.

2.0 DESCRIPTION OF PROPOSED AMENDMENT

The proposed changes include replacement of TS Figures 2-1A (RCS Pressure - Temperature Limits for Heatup) and 2-1B (RCS Pressure - Temperature Limits for Cooldown) with the single composite TS Figure 2-1 (RCS Pressure - Temperature Limits for Heatup, Cooldown, and Inservice Test). Specifically, this new figure applies the use of American Society of Mechanical Engineers (ASME) Code Case N-640 which allows the use of the conservative, but less limiting, lower bound of static initiation critical stress intensity factor (K_{IC}) versus the lower bound of static, dynamic, and crack arrest critical stress intensity factor (K_{IA}). OPPD did not invoke the use of ASME Code Case N-514 in its current low temperature overpressure protection (LTOP) analysis, which allows the P-T limit curve to be exceeded by an additional 10% (Reference 10.3). The current LTOP analysis is applicable to this proposed pressure and temperature (P-T) limit curve. Therefore, Technical Specification 2.3(3) does not need to be revised. Furthermore, OPPD will ensure that any new LTOP analysis will ensure that the setpoint curve will prevent the pressure excursion associated with a limiting LTOP pressurization event from exceeding 100% of this TS Figure. Additionally, the adjusted reference temperatures (ARTs) for the 1/4T (237.76°F) and 3/4T (187.97°F) location have been determined in accordance with Regulatory Guide 1.99, Revision 2, using a predicted fluence value of 2.15×10^{19} n/cm^2 which corresponds to 40 EFPY as stated in Table 6.2-1 of Reference 10.1. This predicted fluence value is applicable to the limiting weld for Fort Calhoun Station for the 3-410 axial welds using weld wire heat number 12008/13253 at the critical 60° location (Reference 10.2). Due to the application of these ART values with its associated predicted fluence value, Technical Specification 2.1.2(6)(a) will be changed to

reflect the new fast neutron ($E \geq 1$ Mev) fluence to 2.15×10^{19} n/cm^2 which corresponds to 40 EFPY. Therefore, this P-T limit curve will be valid to 40 EFPY of reactor operation, which corresponds to approximately April 2022. OPPD is requesting the use of TS Figure 2-1 even though it exceeds the current operating license term based on it being more conservative due to the use of the higher predicted fluence value for the ARTs. 40 EFPY was chosen to be incorporated into TS Figure 2-1 to maximize the available cooldown rates while maintaining a sufficient operating window for the low temperature overpressure protection (LTOP) analysis. Based on the deletion of TS Figures 2-1A and 2-1B and the incorporation of TS Figure 2-1, the following Technical Specifications are required to be changed: 2.1.1(8), 2.1.2, 2.1.2(1), 2.1.2(2), 2.1.2(6), 2.1.2(6)(a), 2.1.2(6)(c), 2.1.2(6)(d), and 2.1.6(4).

OPPD requests the reduction of the Lowest Service Temperature (Technical Specification 2.1.2(6)(c)) from 182°F to 164°F to be in compliance with Reference 10.9, Section III, NB-2332.

Additionally, the basis for the minimum boltup temperature will be changed to be in accordance with Reference 10.4, Section XI, Appendix G. TS Figure 2-1 has been analyzed to a minimum boltup temperature of 64°F, but due to the Low Temperature Overpressure Protection (LTOP) analysis being analyzed to 82°F, the minimum boltup temperature will remain at 82°F at this time. As a result of shifting to Reference 10.4, Section XI, Appendix G, Technical Specification 2.1.2(6)(c) will be changed to reflect a RT_{NDT} versus NDTT.

Finally, OPPD proposes the following associated revisions to the Basis section of Technical Specification 2.1.2: (1) update Reference 2 to reflect only the ASME code; (2) update the predicted fluence from 1.5×10^{19} n/cm^2 to 2.15×10^{19} n/cm^2 and state that it is valid to 40 EFPY; (3) revise the RT_{NDT} values at 40 EFPY at the 1/4T and 3/4T location; (4) change the validity of the P-T limit curve from Cycle 22 to Cycle 34 consistent with the predicted fluence projection changes; (5) update the technical basis for the limit lines in TS Figure 2-1 to be reflective of ASME Code Case N-640, and Westinghouse Electric Company/Combustion Engineering's (W/CE's) P-T limit curve methodology; (6) update the pressure correction factors, pressure and temperature instrumentation uncertainty; (7) update where pressure instrumentation uncertainty is applied; (8) update the technical basis and the value for the lowest service temperature; (9) update the technical basis for the minimum boltup temperature; (10) update to reflect the deletion of TS Figures 2-1A and 2-1B and application of TS Figure 2-1; (11) add Reference 10.7 which reflects NRC approval of W/CE's P-T limit curve methodology; and (12) update text to reflect use of Reference 2, ASME Code Case N-640, and W/CE's P-T limit curve methodology.

In Summary, the proposed amendment replaces TS Figures 2-1A (RCS Pressure - Temperature Limits for Heatup) and 2-1B (RCS Pressure - Temperature Limits for Cooldown) with the single TS Figure 2-1 (RCS Pressure - Temperature Limits for Heatup and Cooldown). Additionally, it changes the lowest service temperature to 164°F and the basis for the minimum boltup temperature. The Basis section for Technical Specification

2.1.2 is being updated to reflect these changes as well as the use of ASME Code Case N-640, and W/CE's P-T limit curve methodology as applicable. Finally, based on the deletion of TS Figures 2-1A and 2-1B and the incorporation of TS Figure 2-1, the following Technical Specifications are required to be changed: 2.1.1(8), 2.1.2, 2.1.2(1), 2.1.2(2), 2.1.2(6), 2.1.2(6)(a), 2.1.2(6)(c), 2.1.2(6)(d), and 2.1.6(4).

3.0 BACKGROUND

Fort Calhoun Station will be conducting a spring 2002 refueling outage. Forty (40) operating fuel assemblies will be replaced with new assemblies. Replacement of these 40 fuel assemblies should result in efficient, defect-free core operation.

The Fort Calhoun fuel integrity, status, and corrective actions have frequently been communicated to the NRC by OPPD through public meetings and submittals of Monthly Operating Reports and periodic performance indicator data.

OPPD plans to implement the P-T Limits Report process that is approved and described by the NRC in Reference 10.7. However, there isn't sufficient time for the submittal and approval of the removal of the P-T limit curves from the Technical Specifications and incorporating them into a RCS P-T Limit Report (PTLR) prior to this new refueling outage date. Therefore, to take advantage of the higher cooldown rates that are available through the application of this new TS Figure, specifically for this refueling outage, this submittal requests the use of this new P-T limit curve based on ASME Code Case N-640 and W/CE's P-T limit curve methodology and associated lowest service temperature. The fluence value used in the development of TS Figure 2-1 was taken from Reference 10.1, Table 6.2-1 for 40 EFPY at the 60° location. This fluence value is applicable for 40 EFPY while the fluence value used in Reference 10.11 used the same Table and location but was applicable to 24.25 EFPY. Reference 10.1 was reviewed and approved in References 10.2 and 10.11.

4.0 REGULATORY REQUIREMENTS & GUIDANCE

The regulations that govern the development and application into the Technical Specifications of the P-T limit curve are 10 CFR 50.36(c)(2); Appendices G and H of 10 CFR 50; Generic Letter 88-11; Regulatory Guide 1.99, Revision 2; and Standard Review Plan Section 5.3.2. The regulation that governs the basis and value for the minimum boltup temperature is 10 CFR 50 Appendix G. The lowest service temperature is governed by 10 CFR 50 Appendix G and Reference 10.9, Section III, NB 2332.

5.0 TECHNICAL ANALYSIS

5.1 Design Basis

In accordance with Reference 10.4, Section XI, Appendix G, the minimum boltup temperature should be at least equal to the initial RT_{NDT} temperature for the material in the stressed regions plus any effects of irradiation at the stressed regions. Additionally, temperature instrumentation uncertainty is conservatively applied. Currently, the basis for the minimum boltup temperature is a measured 10°F NDTT of the reactor vessel flange, which is not subject to radiation damage, plus 60°F data scatter in NDTT measurements, plus 12°F instrument error. An evaluation (Reference 10.5) was performed by Westinghouse Electric Company (WEC) that determined that the limiting component of the flange region materials were the shell plate (D-4801-3) and the closure head torus (D-4808). The RT_{NDT} of these components was 10°F. The temperature uncertainty to be used in this calculation is 14°F which bounds the instruments in the control room that operators would use to determine reactor coolant system (RCS) temperature. Therefore, since this area is not subject to radiation damage, the minimum boltup temperature is determined in accordance with Reference 10.4, Section XI, Appendix G as follows:

$$\text{Minimum boltup temperature} = \text{Initial } RT_{NDT} + \text{Temperature Uncertainty}$$

$$= 10°F + 14°F = 24°F$$

For this temperature to be used, the P-T limit curve and the LTOP analysis must both be analyzed to that temperature. TS Figure 2-1 has been conservatively analyzed to an indicated temperature of 64°F. The FCS LTOP analysis is only analyzed to an indicated temperature of 82°F. Therefore, OPPD proposes to change the technical basis for the minimum boltup temperature in Technical Specification 2.1.2(6)(c) and the Basis section for Technical Specification 2.1.2, but maintain the minimum boltup temperature at 82°F until a revised LTOP analysis is submitted to the NRC for review and approval.

The lowest service temperature is defined by Reference 10.9, Section III, NB-2332 as the Initial RT_{NDT} for piping and material for pumps, valves, and fittings with any pipe connections + 100°F. In addition, a temperature instrumentation uncertainty of 14°F will be applied. This temperature uncertainty conservatively bounds those instruments in the control room that Operators will use to determine RCS temperature. The primary system components for Fort Calhoun were designed and fabricated before the ASME Code, Section III, NB 2300 had been instituted. NB 2300 established the Code requirements for determination of RT_{NDT} in the Summer 1972 Code Addenda. The basis for the 50°F maximum RT_{NDT} for piping, pumps, and valves upon which the lowest service temperature is established is a conservative estimate based on shop experience and

fabrication specifications for later plants. (Reference 10.5) The lowest service temperature is determined as follows:

Lowest Service Temperature = Initial RT_{NDT} + 100°F + temperature uncertainty

= 50°F + 100°F + 14°F = 164°F

Therefore, a lowest service temperature of 164°F is proposed to take the place of the conservative lowest service temperature value of 182°F that is currently located in Technical Specification 2.1.6(c) and the Basis section for Technical Specification 2.1.2.

Per 10 CFR 50 Appendix G, before pressure can exceed 20% of the pre-service hydrostatic test pressure, the controlling material must be attained. The controlling material is either the material in closure flange region or the material in the beltline region with the highest reference temperature, depending on the operating condition. The limiting flange limit is the one for normal operation and is determined in accordance with 10 CFR 50 Appendix G as follows:

Flange Limit $_{\text{NORMAL OP}}$ = Initial RT_{NDT} + 120°F + temperature uncertainty

= 10°F + 120°F + 14°F = 144°F

The minimum pressure is defined by 10 CFR 50 Appendix G as 20% of the pre-service hydrostatic test pressure and is determined as follow:

20% of Preservice Hydrostatic Test Pressure = (1.25 x Design Pressure) x 0.20

= 1.25 x 2500 psi x 0.20 = 625 psia

The pressure correction factors due to flow and elevation have been determined by WEC as the following (Reference 10.5):

RCS temperature < 210°F = 61 psi
RCS temperature $\geq$ 210°F = 67 psi

These values are updated from the previous values stated in Technical Specification 2.1.2 Basis. They take into account the removal of the steam generator orifice plates (resulting in slightly higher RCS flow and corresponding changes in the hydraulic pressure drop component. Note: Higher RCS flow is more conservative) and due to a lower value for minimum bolt-up temperature (affecting the density compensation in the elevation head component. Note: The minimum boltup temperature that the pressure correction factors were analyzed to is 64°F, which includes temperature instrumentation uncertainty.)

Therefore, the minimum pressure would be 564 psia for RCS temperatures < 210°F, and 558 psia for RCS temperatures $\geq$ 210°F. Additionally, these pressure correction factors are incorporated into the limit lines of TS Figure 2-1.

Hence, the lowest service temperature is the most limiting temperature before pressure can exceed 20% of the preservice hydrostatic test pressure, which corresponds to 564 psia.

The LTOP-enable temperatures for both the heatup and cooldown condition are stated as follows (Reference 10.5):

LTOP-Enable Temperature (Cooldown) = 301.76°F
LTOP-Enable Temperature (Heatup) = 324.0°F

This is the value for a heatup rate of 75°F/Hr. This value corresponds to the heatup rate limit that is used in the P-T limit curve.

The LTOP-enable temperature is sensitive to the P-T limit curve only due to where pressure instrument uncertainty is applied. Above the LTOP-enable temperature, pressure instrument uncertainty is incorporated into the P-T limit curves and below this temperature it is not. A conservative LTOP-enable temperature of 350°F was used in the development of the P-T limit curve. This value was chosen to bound both the heatup and cooldown LTOP-enable temperatures and to optimize the operating margin between the P-T limit curve and the LTOP analysis setpoint curve. Currently, the FCS LTOP-enable temperature is 385°F and will stay at this temperature until the LTOP analysis is revised. Therefore, some conservatism in the P-T limit curve will exist (due to the addition of pressure instrument uncertainty between 350°F and 385°F) until the LTOP analysis is revised to use this lower LTOP enable temperature. (Note: Pressure instrument uncertainty is not applied below the LTOP enable temperature due to it being incorporated into the LTOP analysis setpoint curve). A pressure instrumentation uncertainty of 50 psi is being used, which bounds the wide and narrow range pressurizer pressure instruments.

The ARTs used in TS Figure 2-1 were calculated in accordance with Regulatory Guide 1.99, Revision 2, Regulatory Position 1.1 (Reference 10.6). They use a predicted base metal surface fluence value of 2.15 x 10^{19} n/cm^2 (Reference 10.1). This value corresponds to the predicted fluence value associated with 40 EFPY of reactor operation to the 60° location. The 60° location corresponds to the critical 3-410 axial weld for tandem weld wire heat 12008/13253 (Reference 10.2). The ARTs that were used in TS Figure 2-1 are as follows:

1) ART value at the clad to reactor vessel base metal-clad interface is 261.56°F.
2) ART value for the 1/4T position is 237.76°F.
3) ART value for the 3/4 T position is 187.97°F.

TS Figure 2-1 was developed using W/CE's methodology for P-T curve development (Reference 10.5), Reference 10.4, Section XI, Appendix A and G, ASME Code Case N-640, and 10 CFR 50 Appendix G. W/CE's methodology was reviewed and approved in References 10.7 and 10.8. TS Figure 2-1 has incorporated the following information:

1) A conservative indicated minimum boltup temperature of 82°F. TS Figure 2-1 has been analyzed to an indicated temperature of 64°F that will be requested to be used when the LTOP analysis is re-performed and submitted to the NRC for review.
2) An indicated lowest service temperature of 164°F.
3) An actual ART value at the clad to reactor vessel base metal interface is 261.56°F.
4) An actual ART value for the 1/4T position is 237.76°F.
5) An actual ART value for the 3/4 T position is 187.97°F.
6) An indicated LTOP-enable temperature at 350°F.
7) Pressure correction factors due to flow and elevation:
   An actual RCS temperature < 210°F = 61 psi
   An actual RCS temperature > 210°F = 67 psi
8) Pressure instrumentation uncertainty above an indicated RCS temperature of 350°F of 50 psi.
9) Temperature instrumentation uncertainty of 14°F.
10) A minimum pressure value of 564 psia for actual RCS temperatures < 210°F. (Note: The endpoint of the minimum pressure line corresponds to the lowest service temperature value of 164°F).
11) The current LTOP analysis setpoints have been evaluated by WEC (Reference 10.3). They have been determined to be bounding and applicable to TS Figure 2-1. Therefore, Technical Specification 2.3(3) does not need to be revised. OPPD did not invoke ASME Code Case N-514 and limited LTOP overpressurization events to 100% of the previous P-T limits shown in TS Figures 2-1A and 2-1B. When the LTOP analysis is re-performed, OPPD will limit the LTOP overpressurization events to 100% of the P-T limit lines of TS Figure 2-1.
12) TS Figure 2-1 is a composite P-T limit curve that is a conservative blend of the limiting heatup or cooldown rate. The following heatup and cooldown rates must be followed to maintain the validity of TS Figure 2-1:

    The heatup rate is limited to 75°F/hr for an indicated RCS temperature of 82°F and greater.
    The cooldown rate is limited to 100°F/hr for an RCS temperature greater than or equal to 178°F. Below 178°F, it is limited to 50°F/hr.

13) The hydrostatic pressure and leak test curve was developed using the same techniques as the heatup and cooldown limit curve with the exception that a safety factor of 1.5 is allowed by Reference 10.4, Section XI, Appendix G.

The data table that was used to plot the composite heatup and cooldown limit lines for TS Figure 2-1 is located in the Attachment to Reference 10.3, Table 1 for Composite Curve. The data point at 223.9°F was conservatively not plotted. This data point is due to the shift in pressure correction factors. The heatup and cooldown limit lines that are plotted are more conservative below the lowest service temperature (164°F) than the minimum pressure (564 psia) which is required by 10 CFR 50 Appendix G.

The data table that was used to plot the Hydrostatic Pressure and Leak Test curve is located in Reference 10.5, Table 3 for isothermal conditions. For temperatures below the lowest service temperature (164°F), this curve can not exceed the minimum pressure (564 psia); hence below 164°F, this curve is plotted within these 10 CFR 50 Appendix G regulatory requirements. The data point at 223.9°F was conservatively not plotted. This data point is due to the shift in pressure correction factors. The data point at 349.9°F was conservatively not plotted. This data point is due to the addition of pressure instrumentation uncertainty at 350°F and above.

5.2 Risk Information

The proposed amendment does not involve application or use of risk-informed decisions. The risk to the health and safety of the public as a result of a LTOP and subsequent reactor vessel failure is minimal due to implementation of these changes.

6.0 REGULATORY ANALYSIS

The proposed amendment to change the P-T limit curve conforms to 10 CFR 50.36(c)(2); Appendices G and H of 10 CFR 50; Generic Letter 88-11; Regulatory Guide 1.99, Revision 2; and Standard Review Plan Section 5.3.2. The technical basis for TS Figure 2-1 is possible due to approval of an exemption to 10 CFR 50.60(b). This exemption allows the use of ASME Code Case N-640, which permits an administrative shift in the reference stress intensity factor from K_{IA} to K_{IC}. The use of K_{IC} as the reference stress intensity factor, while less limiting, is still conservative in nature. Additionally, the ARTs are being updated in accordance with Regulatory Guide 1.99, Revision 2. Therefore, there is no reduction to the margin of safety. The lowest service temperature is an administrative requirement that is being updated to be in compliance with Reference 10.9, Section III, NB-2332. Therefore, there is no reduction to the margin of safety. The shift in the basis for the minimum boltup temperature from NDTT to RD_{NDT} is an administrative requirement that is being updated to in compliance with Reference 10.4, Section XI, Appendix G. Therefore, there is no reduction to the margin of safety.

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

7.0 NO SIGNIFICANT HAZARDS CONSIDERATION

OPPD has evaluated whether or not a significant hazards consideration is involved with the proposed amendment(s) by focusing on the three standards set forth in 10 CFR 50.92, "Issuance of Amendment," as discussed below:

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

   Response: No.

   The proposed changes will not increase the probability or consequence of any accident for the following reasons:

   1) TS Figure 2-1 is proposed to incorporate the use of ASME Code Case N-640, which has been approved by the NRC as being acceptable for the development of P-T curves. Additionally, it is being updated for operation to higher neutron fluence values for use in the ART calculations.
   2) Reducing the lowest service temperature is in compliance with Reference 10.9, Section III, NB-2332.
   3) The shift in the basis for minimum boltup temperature is in compliance with 10 CFR 50 Appendix G.
   4) Updating the fluence and EFPY applicability is in compliance with Regulatory Guide 1.99, Revision 2.

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

   Response: No.

   The proposed revision does not change any equipment required to mitigate the consequences of an accident. The continued use of the same Technical Specification administrative controls prevents the possibility of a new or different kind of accident. Since the proposed changes do not involve the addition or modification of equipment nor alter the design of plant systems, the proposed changes do not create the possibility of a new or different kind of accident from any accident previously evaluated. The changes proposed do not change how design basis accident events are postulated nor do the changes themselves initiate a new kind of accident or failure mode with a unique set of conditions (proposed administrative controls). Therefore, the proposed change does not create the possibility of a new or different kind of accident from any previously evaluated.

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

Response: No.

The proposed TS Figure 2-1 does not constitute a significant reduction in the margin of safety due to the following:

1) The current LTOP analysis setpoints are bounding and applicable to this TS Figure.
2) The use of ASME Code Case N-640 has been approved by the NRC as acceptable for the development of P-T limit curves. W/CE's P-T limit curve methodology has been approved for the development of P-T curves.
3) The reduction in lowest service temperature is in compliance with Reference 10.9, Section III, NB-2332.
4) The shift in the basis of the minimum boltup temperature from NDTT to RT_{NDT} is in compliance with Reference 10.4, Section XI, Appendix G.
5) Updating the fluence and EFPY applicability of the TS Figure 2-1 to maintain validity is in compliance with Regulatory Guide 1.99, Revision 2.

The P-T curve results, with the proposed changes, remain within the regulatory acceptance criteria utilizing W/CE methodology and ASME Code Case N-640. These criteria, 10 CFR 50.36(c)(2), have been developed for application to analyses performed for long term operation of reactor vessels. These acceptance criteria have been developed for the purpose of use in design basis accident analyses such that meeting these limits demonstrates adequate protection of public health and safety. An acceptable margin of safety is inherent in these licensing limits. Therefore, the proposed changes do not involve a reduction in a margin of safety.

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

8.0 ENVIRONMENTAL CONSIDERATION

The proposed amendment is confined to administrative procedures or requirements. OPPD bases these changes on the revised P-T limit curve calculations. The changes meet the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9) for the following reasons:

1) As demonstrated in Section 7.0, the proposed amendment does not involve a significant hazards consideration.
2) The proposed amendment does not result in a significant change in the types or increase in the amounts of any effluents that may be released offsite. Also, the TS change does not introduce any new effluents or significantly increase the quantities of existing effluents. As such, the change cannot significantly affect the types or amounts of any effluents that may be released offsite.

3) The proposed amendment does not result in a significant increase in individual or cumulative occupational radiation exposure. The proposed change does not result in any physical plant changes. No new surveillance requirements are anticipated as a result of these changes that would require additional personnel entry into radiation controlled areas. Therefore, the amendment has no significant affect on either individual or cumulative occupational radiation exposure.

Accordingly, the proposed amendment meets the eligibility criterion for categorical exclusion set forth in 10 CFR 51.22(c)(9). Therefore, pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the proposed amendment.

9.0 PRECEDENCE

The NRC previously reviewed and approved the use of W/CE's P-T limit curve methodology in References 10.7 and 10.8.

The NRC has previously reviewed and approved relief exemptions for the use of ASME Code Case N-640. (Reference 10.10)

The NRC endorses via 10 CFR 50 Appendix G and 10 CFR 50.55a the ASME Boiler and Pressure Vessel Code, 1995 Edition and addenda through the 1996 Addenda Section XI and Section III, respectively.

10.0 REFERENCES

10.1 WCAP-15443, Revision 0, "Fast Neutron Fluence Evaluations for the Fort Calhoun Unit 1 Reactor Pressure Vessel," dated July 2000
10.2 Letter from NRC (A. B. Wang) to OPPD (S. K. Gambhir), dated June 6, 2001, "Fort Calhoun Station Unit No. 1 - Issuance of Amendment - Deletion of Section 3.D, "License Term" (TAC No. MA9690)" (NRC-01-058)
10.3 Letter LTR-PS-01-26, Revision 0, from WEC (F. P. Ferraraccio) to OPPD (K. Holthaus), "Evaluation of the Current LTOP Analysis for Revised Technical Specification P-T Limits at the Fort Calhoun Station," dated November 7, 2001 (Located in Attachment 3)
10.4 ASME Boiler and Pressure Vessel Code, 1995 Edition and Addenda through the 1996 Addenda
10.5 Report DAR-PS-01-4, Revision 0, "Reactor Coolant System Pressure-Temperature Limits and Low Temperature Overpressure Enable Temperature for 40 Effective Full Power Years for Fort Calhoun Station Unit 1" (Located in Attachment 3)
10.6 FC06799, Revision 0, "40 EFPY Pressure and Temperature Limit Curve Inputs"
10.7 Safety Evaluation of Topical Report CE NPSD-683, Revision 6, "Development of a RCS Pressure and Temperature Limits Report (PTLR) for the Removal of P-T

Limits and LTOP Requirements from the Technical Specifications" (TAC No. MA9561)

10.8 Letter from G. F. Wunder, Project Manager, U.S. Nuclear Regulatory Commission, to J. Knubel, Senior Vice President, New York Power Authority, "Exemption from the Requirements of 10CFR 50.60, 'Acceptance Criteria for Fracture Prevention Measures for Lightwater Nuclear Power Reactors for Normal Operation,' to Allow for Use of Alternative Methodology for Construction of Pressure Temperature Limit Curves - Indian Point Nuclear Generating Unit No. 3 (TAC No. M99928)," April 10, 1998

10.9 ASME Boiler and Pressure Vessel Code, 1986 Edition

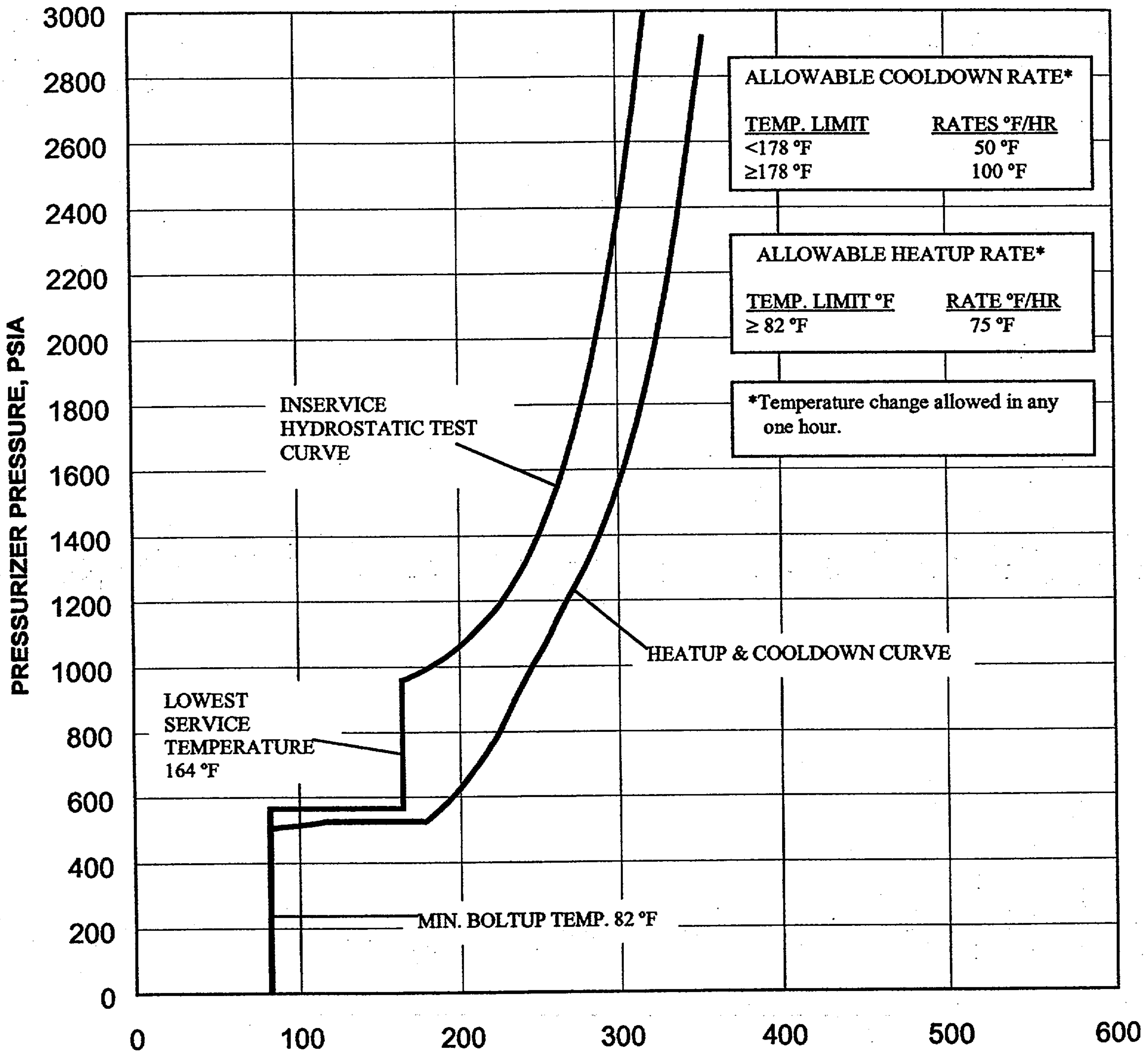
10.10 Letter from Donna Skay, Project Manager, U.S. Nuclear Regulatory Commission, to C. Cruse, Vice President-Nuclear Energy, Calvert Cliffs Nuclear Power Plant, Inc., "Calvert Cliffs Nuclear Power Plant, Unit Nos. 1 and 2 - Amendment Re: Pressure-Temperature Curves (TAC Nos. MA9999 and MB0000)," dated March 15, 2001

10.11 Letter from NRC (L. R. Wharton) to OPPD (S. K. Gambhir), dated March 28, 2001, "Fort Calhoun Station, Unit No. 1 - Issuance of Amendment Re: Extending Pressure-Temperature (P-T) Limits (TAC No. MB0322)" (NRC-01-030)

Mark-up of Technical Specifications

Figure 2-1

FORT CALHOUN STATION UNIT 1 COMPOSITE P/T LIMITS, 40 EFPY



RCS Pressure-Temperature Limits for Heatup, Cooldown, and Inservice Test	Omaha Public Power District Fort Calhoun Station-Unit No. 1

TECHNICAL SPECIFICATIONS - FIGURES

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2.0 **LIMITING CONDITIONS FOR OPERATION**
2.1 Reactor Coolant System (Continued)
2.1.1 Operable Components (Continued)

(5) DELETED

(6) Both steam generators shall be filled above the low steam generator water level trip set point and available to remove decay heat whenever the average temperature of the reactor coolant is above 300°F. Each steam generator shall be demonstrated operable by performance of the inservice inspection program specified in Section 3.17 prior to exceeding a reactor coolant temperature of 300°F.

(7) Maximum reactor coolant system hydrostatic test pressure shall be 3125 psia. A maximum of 10 cycles of 3125 psia hydrostatic tests are allowed.

(8) Reactor coolant system leak and hydrostatic test shall be conducted within the limitations of Figure~~s~~ 2-1~~A and 2-1B~~.

(9) Maximum secondary hydrostatic test pressure shall not exceed 1250 psia. A minimum measured temperature of 73°F is required. Only 10 cycles are permitted.

(10) Maximum steam generator steam side leak test pressure shall not exceed 1000 psia. A minimum measured temperature of 73°F is required.

(11) If no reactor coolant pumps are operating, a non-operating reactor coolant pump shall not be started while T_c is below 385°F unless at least one of the following conditions is met:

2.0 LIMITING CONDITIONS FOR OPERATION

2.1 Reactor Coolant System (Continued)

2.1.2 Heatup and Cooldown Rate

Applicability

Applies to the temperature change rates and pressure of the reactor coolant system.

Objective

To specify limiting conditions of the reactor coolant system heatup and cooldown rates.

Specification

The reactor coolant pressure shall be limited during plant operation in accordance with Figure 2-1~~A and 2-1B~~ and as follows:

(1) Allowable combinations of pressure and temperature (T_c) for a specific heatup rate shall be below and to the right of the applicable limit lines as shown on Figure 2-1~~A~~.

(2) Allowable combinations of pressure and temperature (T_c) for a specific cooldown rate shall be below and to the right of the applicable limit lines as shown on Figures 2-1B.

(3) The heatup rate of the pressurizer shall not exceed l00°F in any one hour period.

(4) The cooldown rate of the pressurizer shall not exceed 200°F in any one hour period.

(5) When any of the above limits are exceeded, the following corrective actions shall be taken:

(a) Immediately initiate action to restore the temperature or pressure to within the limit.

(b) Perform an analysis to determine the effects of the out of limit condition on the fracture toughness properties of the reactor coolant system.

(c) Determine that the reactor coolant system remains acceptable for continued operation or be in cold shutdown within 36 hours.

(6) Before the radiation exposure of the reactor vessel exceeds the exposure for which they apply, Figures 2-1~~A and 2-1B~~ shall be updated in accordance with the following criteria and procedures:

2.0 LIMITING CONDITIONS FOR OPERATION
2.1 Reactor Coolant System (Continued)
2.1.2 Heatup and Cooldown Rate (Continued)

(a) Figure~~s~~ 2-1~~A and 2-1B are~~ is valid for a fast neutron ($E \geq 1$MeV) fluence of 2.15 ~~1.50~~X10^{19} n/cm^2 which corresponds to 40 ~~24.25~~ EFPY.

(b) The limit line on the figure~~s~~ shall be updated for a new integrated power period as follows: the total integrated reactor thermal power from startup to the end of the new period shall be converted to an equivalent integrated fast neutron exposure ($E \geq 1$ MeV).

(c) The limit lines in Figure~~s~~ 2-1~~A and 2-1B~~ shall be moved parallel to the temperature axis (horizontal) in the direction of increasing temperature a distance equivalent to the transition temperature shift during the period since the curves were last constructed. The boltup temperature limit line shall remain at 82°F as it is set by the RT_{NDT} ~~NDTT~~ of the reactor vessel flange and not subject to fast neutron flux. The lowest service temperature shall remain at 164 ~~182~~°F because components related to this temperature are also not subject to fast neutron flux.

(d) The Technical Specification 2.3(3) shall be revised each time the curves of Figure~~s~~ 2-1~~A and 2-1B~~ are revised.

Basis

All components in the reactor coolant system are designed to withstand the effects of cyclic loads due to reactor coolant system temperature and pressure changes.[(1)] These cyclic loads are introduced by normal unit load transients, reactor trips and startup and shutdown operation.

During unit startup and shutdown, the rates of temperature and pressure changes are limited. The design number of cycles for heatup and cooldown is based upon allowable heatup/cooldown rates and cyclic operation.

2.0 LIMITING CONDITIONS FOR OPERATION
2.1 Reactor Coolant System (Continued)
2.1.2 Heatup and Cooldown Rate (Continued)

The maximum allowable reactor coolant system pressure at any temperature is based upon the stress limitations for brittle fracture considerations. These limitations are derived by using the rules contained in Section XI ~~III~~[2] of the ASME Code including Appendix A and G, ~~Protection Against Nonductile Failure,~~ Westinghouse Electric Company/Combustion Engineering's P-T (W/CE's) limit curve methodology[11], and the rules contained in 10 CFR 50, Appendix G, Fracture Toughness Requirements. This ASME Code assumes that a crack 10-11/16 inches long and 1-25/32 inches deep exists on the inner surface of the vessel. Furthermore, operating limits on pressure and temperature assure that the crack does not grow during heatups and cooldowns.

The reactor vessel beltline material consists of six plates. The nilductility transition temperature (T_{NDT}) of each plate was established by drop weight tests. Charpy tests were then performed to determine at what temperature the plates exhibited 50 ft-lbs. absorbed energy and 35 mils lateral expansion for the longitudinal direction. NRC technical position MTEB-5-2 was used to establish a reference temperature for transverse direction (RT_{NDT}) of -12°F.

The initial RT_{NDT} value for the Fort Calhoun submerged arc vessel weldments was determined to be -56°F with a standard deviation of 17°F. By applying the shift prediction methodology of Regulatory Guide 1.99, Revision 2, a weld material adjusted reference temperature (RT_{NDT}) was established at 10°F based on the mean value plus two standard deviations. The standard deviation was determined by using the root-mean-squares method to combine the margin of 28°F for uncertainty in the shift equation with the margin of 17°F for uncertainty in the initial RT_{NDT} value.

Similar testing was not performed on all remaining material in the reactor coolant system. However, sufficient impact testing was performed to meet appropriate design code requirements[3] and a conservative RT_{NDT} of 50°F has been established.

As a result of fast neutron irradiation in the region of the core, there will be an increase in the T_{NDT} with operation. The techniques used to predict the integrated fast neutron ($E \geq 1$ MeV) fluxes of the reactor vessel are described in Reference 5 with the result that the integrated fast neutron flux ($E \geq 1$ MeV) is 1.73×10^{19} n/cm^2, including tolerance at the inside surface of the critical reactor vessel beltline weld material, over the 40 years design life of the vessel.

Since the neutron spectra and the flux measured at the samples and reactor vessel inside radius should be nearly identical, the measured transition shift for a sample can be applied to the adjacent section of the reactor vessel for later stages in plant life equivalent to the difference in calculated flux magnitude. The maximum exposure of the reactor vessel will be obtained from the measured sample exposure by application of the calibrated azimuthal neutron flux variation. The maximum integrated fast neutron ($E \geq 1$ MeV) exposure of the reactor vessel at the critical reactor vessel beltline location including tolerance is computed to be 1.73×10^{19} n/cm^2 at the vessel inside surface for 40 years operation at

2.0 LIMITING CONDITIONS FOR OPERATION

2.1 Reactor Coolant System (Continued)

2.1.2 Heatup and Cooldown Rate (Continued)

1500 MWt and 85% load factor. The predicted shift at this location at the 1/4t depth from the inner surface is projected to be 252°F, including margin, using the shift prediction equation of Regulatory Guide 1.99, Revision 2. The actual shift in T_{NDT} will be re-established periodically during the plant operation by testing of reactor vessel material samples which are irradiated cumulatively by securing them near the inside wall of the reactor vessel as described in Section 4.5.3 and Figure 4.5-1 of the USAR. To compensate for any increase in the T_{NDT} caused by irradiation, limits on the pressure-temperature relationship are periodically changed to stay within the stress limits during heatup and cooldown. Analysis of the three removed irradiated reactor vessel surveillance specimens[8,9 and 10], combined with weld chemical composition data and implementation of extreme low radial leakage core loading designs beginning in Cycle 14, indicate that the fluence at the end of 40 ~~24.25~~ Effective Full Power Years (EFPY) at 1500 MWt will be 2.15 ~~1.50~~x10^{19} n/cm^2 on the inside surface of the reactor vessel. This results in a total shift of the RT_{NDT} of 237.76 ~~238.5~~°F, including margin, for the area of greatest sensitivity (weld metal) at the 1/4t location using Regulatory Guide 1.99, Revision 2, and a shift of 187.97 ~~187.5~~°F at the 3/4t location. Operation through fuel Cycle 34 ~~22~~ will result in less than 40 ~~24.25~~ EFPY.

The limit lines in Figure 2-1 are based on Reference 2, Appendix G, W/CE's methodology for P-T limit curve generation[11], and ASME Code Case N-640 as discussed below.

Reference Stress Intensity Factor

The reference stress intensity factor (K_{IR}) used in the development of the limit lines in Figure 2-1 is based on ASME Code Case N-640. This Code case allows the use of K_{Ic} (lower bound of static initiation critical stress intensity factor) and is an approved exemption by the NRC in accordance with 10CFR50.60(b). K_{Ic} is obtained from a reference fracture toughness curve for reactor pressure vessel low alloy steels as defined in Appendix A to Section XI of the ASME Code and is approximated by the following equation:

$$K_{Ic} = 33.20 + 20.734e^{[0.0200(T-RT_{NDT})]}$$

where,

K_{Ic} = Crack initiation reference stress intensity factor, Ksi $\sqrt{in}$

T = temperature at the postulated crack tip, °F

RT_{NDT} = adjusted reference nil ductility temperature (also called ART) at the postulated crack tip, °F

For any instant during the postulated heatup or cooldown, K_{Ic} is calculated using the metal temperature at the tip of the flaw, as well as the value of ART at that flaw location.

2.0 LIMITING CONDITIONS FOR OPERATION

2.1 Reactor Coolant System (Continued)

2.1.2 Heatup and Cooldown Rate (Continued)

Regulatory Requirements

The Reference 2, Appendix G equation relating K_{IM}, K_{IT}, and K_{IR} is re-arranged as shown below to solve for the allowable pressure stress intensity factor, K_{IM}, as a function of time with the calculated K_{IR} and K_{IT} values to determine the allowable pressure stress intensity factor and consequently the allowable pressure:

(1) For Service Level A and B operation:

$$K_{IM} = \frac{K_{IR} - K_{IT}}{2}$$

and

(2) For Hydrostatic and Test Conditions when the core is not critical and tests are performed at isothermal conditions (i.e. thermal stress intensity factor, K_{IT}, = 0)

$$K_{IM} = \frac{K_{IR}}{1.5}$$

where,

K_{IM} = Allowable pressure stress intensity factor based on coolant temperature, $Ksi\sqrt{in}$

K_{IR} = Reference stress intensity factor based on coolant temperature, $Ksi\sqrt{in}$

K_{IT} = Thermal stress intensity factor based on coolant temperature, $Ksi\sqrt{in}$

Calculational of P-Allowable

To develop P-T limits, the reactor vessel (RV) beltline region is the only location that receives sufficient neutron fluence to substantially alter the fracture toughness of the RV material. Hence, the beltline region is the most limiting with respect to allowable pressure at any specific temperature. This reduction in fracture toughness is determined using an adjusted reference temperature (ART), which is calculated in accordance with Regulatory Guide 1.99 Revision 2. The allowable pressure is based on the highest ART. The RV beltline region is analyzed assuming a semi-elliptical surface flaw oriented in the axial direction with a depth of one quarter of the RV beltline thickness and an aspect ratio of one to six. This postulated flaw is analyzed at both the inside diameter location

(referred to as the 1/4t location) and the outside diameter location (referred to as the 3/4t location) to assure the most limiting condition is achieved. K_{IT} is determined in accordance with Reference 2, Appendix G-2214.3. The thermal stress intensity factors are determined from the calculated temperature profile through the beltline wall using thermal influence coefficients specifically generated for this purpose. The method employed uses a polynomial fit of the temperature profile and superposition using influence coefficients to calculate K_{IT}. The influence coefficients are dependent upon the geometrical parameters associated with the postulated defect, the geometry of the reactor vessel beltline region, and the assumed unit loading. These influence coefficients were calculated using a 2-dimensional finite element model of the reactor vessel. The influence coefficients were corrected for three-dimensional effects using Reference 2, Appendix A procedures. The K_{IT} and K_{IM} are calculated at any time point in a transient using influence coefficients generated by applying unit loads on a finite element model of the reactor vessel beltline region. The influence coefficients are calculations of stress intensity factors at the 1/4t and 3/4t crack depth location under the following unit loads:

a. for K_{IM-P}, pressure load of 1 ksi, ksi-$\sqrt{in}$/ksi

b. for K_{IT-L}, linear through-wall gradient with peak temperature of 1°F, ksi-$\sqrt{in}$/°F

c. for K_{IT-Q}, quadratic through-wall gradient with peak temperature of 1°F, ksi-$\sqrt{in}$/°F

d. for K_{IT-C}, cubic through-wall gradient with peak temperature of 1°F, ksi-$\sqrt{in}$/°F

Each stress intensity factor is calculated using a standard quarter point element formulation at the respective crack tips. Since all calculations performed are linear, superposition is then used to scale and combine these influence coefficients as necessary to determine the stress intensity factor for a given temperature profile.

In the case of K_{IT}, the through-wall temperature profiles are then fit to the third order polynomial below:

2.0 LIMITING CONDITIONS FOR OPERATION

2.1 Reactor Coolant System (Continued)

2.1.2 Heatup and Cooldown Rate (Continued)

$$T(x) = C_o + C_L (1\text{-}x/_h) + C_Q (1\text{-}x/_h)2 + C_C (1\text{-}x/_h)3$$

where,

$T(x)$ = Temperature at radial location x from inside wall surface, °F
C_o, C_L, C_Q, C_C = Coefficients in polynomial fit, °F
x = Distance through beltline wall, in
h = Beltline wall thickness, in

The coefficients of this polynomial are then combined through the following equation to calculate K_{IT} at the 1/4t and 3/4t locations.

$$K_{IT-Total} = C_L{}^*K_{IT-L} + C_Q{}^*K_{IT-Q} + C_C{}^*K_{IT-C}$$

To calculate the allowable pressure, P-Allowable, the resultant $K_{IT-Total}$ from above in conjunction with Equation (1) from Reference 2, Appendix G-2215, is described as follows.

for Normal Level A and B loads

$$P-Allowable = \frac{K_{IR} - K_{IT-Total}}{2^*K_{IM-P}}$$

for Hydrostatic and Test Conditions, where Isothermal conditions result in $K_{IT-Total} = 0$

$$P-Allowable = \frac{K_{IR}}{1.5^*K_{IM-P}}$$

The P-T limits developed using the method described above account for the temperature differential between the RV base metal and the reactor coolant bulk fluid temperature only. However, uncertainties for instrumentation error, elevation, and flow induced differential pressure differences between the RV beltline and pressurizer are accounted for as follows:

1) Temperature instrumentation uncertainty of 14°F is applied to the entire temperature range of Figure 2-1.

2.0 LIMITING CONDITIONS FOR OPERATION

2.1 Reactor Coolant System (Continued)

2.1.2 Heatup and Cooldown Rate (Continued)

~~The limit lines in Figures 2-1A and 2-1B are based on the following:~~

~~A. Heatup and Cooldown Curves - From Section III of the ASME Code, Appendix G-2215.~~

$$\cancel{K_{IR} = 2\,K_{IM} + K_{IT}}$$

~~K_{IR} = Allowance stress intensity factor at temperature related to RT_{NDT} (ASME III Figure G 2110.1).~~

~~K_{IM} = Stress intensity factor for membrane stress (pressure). The 2 represents a safety factor of 2 on pressure.~~

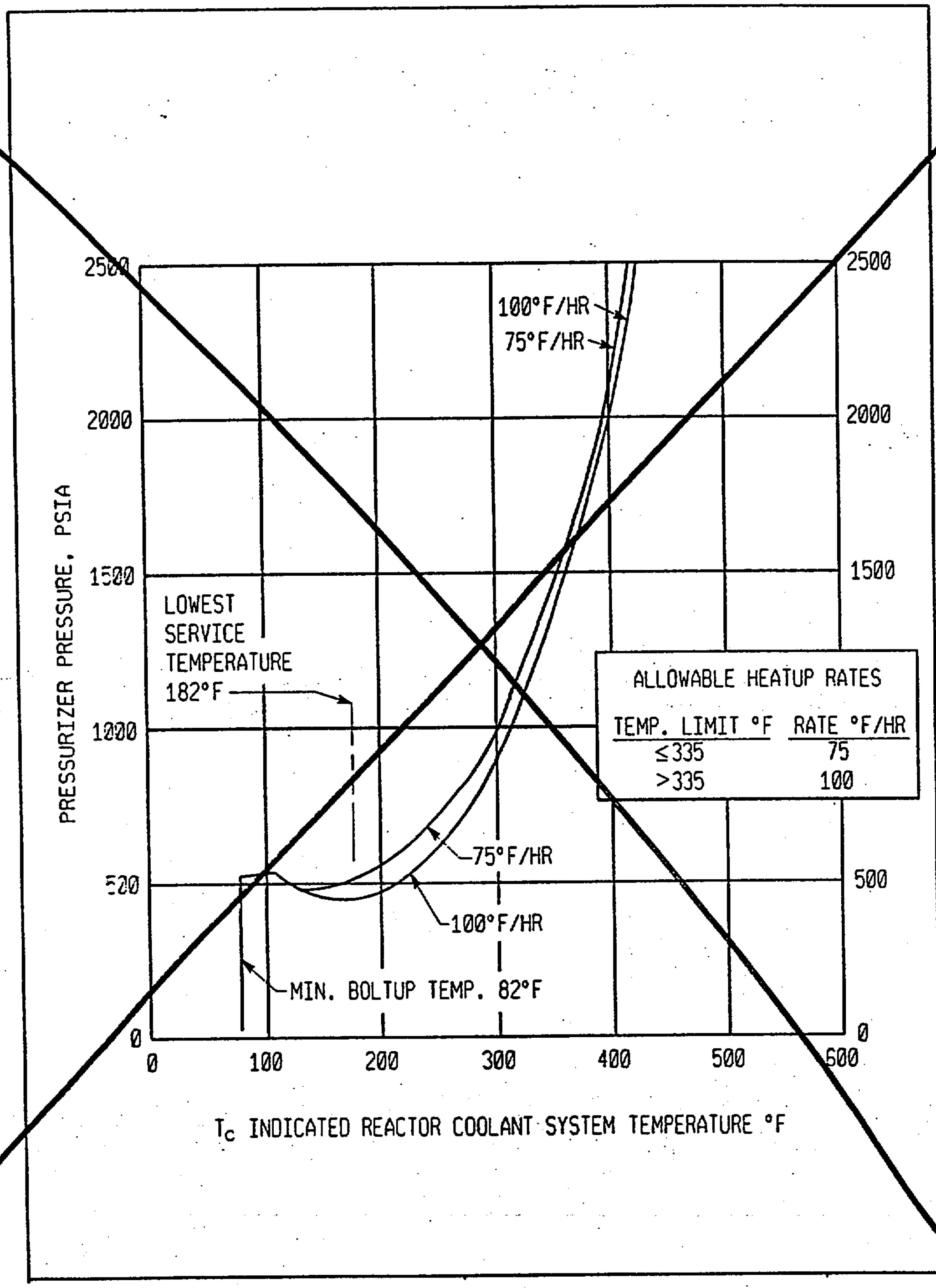
~~K_{IT} = Stress intensity factor radial thermal gradient.~~

~~The above equation is applied to the reactor vessel beltline. For plant heatup the reference stress intensity is calculated for both the 1/4t and 3/4t locations. Composite curves are then generated for each heatup rate by combining the most restrictive pressure-temperature limits over the complete temperature interval.~~

~~For plant cooldown thermal and pressure stress are additive.~~

Figure 2-1A

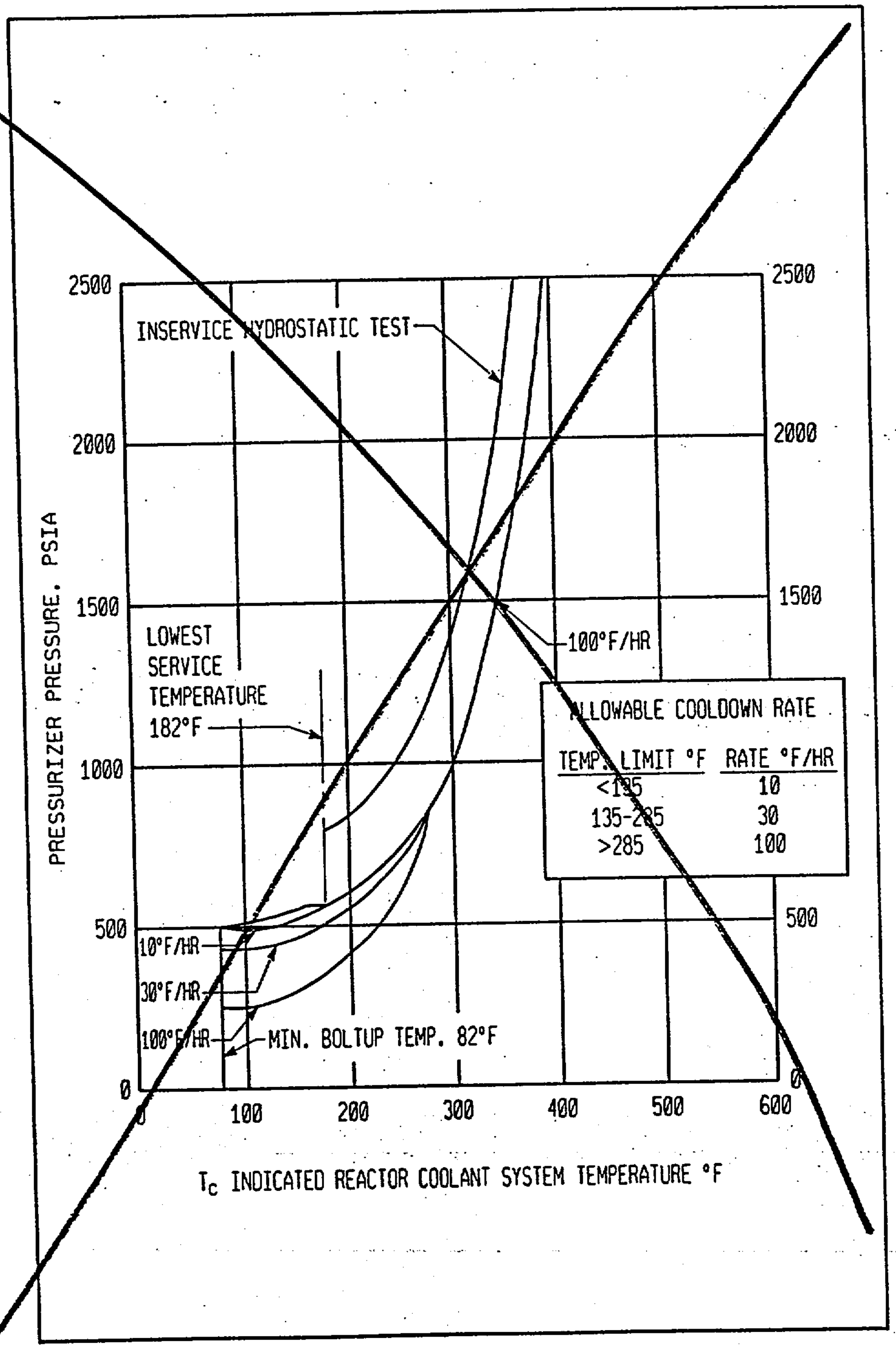
FORT CALHOUN STATION UNIT 1 P/T LIMITS, 24.25 EFPY



RCS Pressure-Temperature Limits for Heatup	Omaha Public Power District Fort Calhoun Station-Unit No. 1

Figure 2-1B

Fort Calhoun Station Unit 1 P/T Limits, 24.25 EFPY



Cooldown and Inservice Test

RCS Pressure-Temperature Limits for Cooldown	Omaha Public Power District Fort Calhoun Station-Unit No. 1

Amendment No. ~~74,77,100,114153, 161, 197~~

2.0 LIMITING CONDITIONS FOR OPERATION

2.1 Reactor Coolant System (Continued)

2.1.2 Heatup and Cooldown Curves (Continued)

$$\cancel{K_{IM} = M_M \frac{PR}{t}}$$

~~M_M = ASME III, Figure G 2214-1~~

~~P = Pressure, psia~~

~~R = Vessel Radius - in.~~

~~t = Vessel Wall Thickness - in.~~

$$\cancel{K_{IT} = MT\Delta T_W}$$

~~MT = ASME III, Figure G-2214-2~~

~~ΔT_W = Highest Radial Temperature Gradient Through Wall at End of Cooldown~~

~~K_{IT} is therefore calculated at a maximum gradient and is considered a constant = A for cooldown and heatup.~~

~~$\frac{M_M R}{t}$ is also a constant = B.~~

~~Therefore:~~

$$\cancel{K_{IR} = AP + B}$$

$$\cancel{P = \frac{K_{IR} - B}{A}}$$

~~K_{IR} is then varied as a function of temperature from Figure G-2110-1 of ASME III and the allowable pressure calculated. Pressure correction factors for elevation and flow (-56 psia for T_c<210°F and -62 psia for $T_c \geq$210°F) and temperature instrumentation uncertainties (+16°F) are considered when plotting the curves. Pressure instrumentation uncertainty is also considered above the LTOP enable temperature of 385°F. Below this temperature, pressure instrumentation uncertainty is accounted for in the LTOP PORV setpoints.~~

2.0 **LIMITING CONDITIONS FOR OPERATION**
2.1 Reactor Coolant System (Continued)
2.1.2 Heatup and Cooldown Curves (Continued)

2) Pressure correction factors that account for the difference in pressure between the reactor vessel beltline and pressurizer pressure instrumentation due to elevation and RCP flow are as follows:
RCS Temperature < 210°F = 61 psi
RCS Temperature > 210°F = 67 psi

3) Below 350°F, pressure instrumentation uncertainty is accounted for in the LTOP system setpoints. Above 350°F, a pressure instrumentation uncertainty of 50 psi is applied to Figure 2-1.

~~B. Inservice Hydrostatic Test - The inservice hydrostatic test curve is developed in the same manner as in A. above with the exception that a safety factor of 1.5 is allowed by ASME III in lieu of 2.~~

~~C.~~ Lowest Service Temperature = 50°F + 100 ~~120~~°F + 14 ~~12~~°F = 164~~182~~°F. As indicated previously, an RT_{NDT} for all material with the ~~exception of the reactor vessel beltline~~ was established at 50°F. Reference 2, Section III, NB-2332 ~~10 CFR Part 50, Appendix G, IV.a.2~~ requires a lowest service temperature of RT_{NDT} + 100 ~~120~~°F for piping, pumps and valves. Below this temperature a pressure of 20 percent of the system hydrostatic test pressure cannot be exceeded. Taking into ~~account pressure correction~~ factors for elevation and flow, this pressure is (.20)(3125) - 61 ~~56~~ = 564 ~~569~~ psia, where 61 ~~56~~ psi is the pressure correction factor ~~hydrostatic head correction factor~~

~~D.~~ Boltup Temperature = 10°F + 14 ~~60°F + 12~~°F = 24 ~~82~~°F. A conservative value of 82°F will be used and maintained. At pressure below 564 ~~569~~ psia, a minimum vessel temperature must be maintained to comply with the manufacturer's specifications for tensioning the vessel head. This temperature is based on RT_{NDT} ~~previous NDTT~~ methods. This temperature corresponds to the measured 10°F RT_{NDT} ~~NDTT~~ of the reactor vessel flange, which is not subject to radiation damage, ~~plus 60°F data scatter in NDTT measurements,~~ plus 14 ~~12~~°F instrument error.

~~E.~~ The temperature at which the heatup and cooldown rates change in Figures 2-1~~A and 2-1B~~ reflects the point at which the most limiting heatup and cooldown rates with respect to the inlet temperature (T_c) change.

2.0 **LIMITING CONDITIONS FOR OPERATION**
2.1 Reactor Coolant System (Continued)
2.1.2 Heatup and Cooldown Rate (Continued)

References:

(1) USAR, Section 4.2.2

(2) ASME Boiler and Pressure Vessel Code, ~~Section III~~

(3) USAR, Section 4.2.4

(4) USAR, Section 3.4.6

(5) WCAP-15443, Revision 0, Fast Neutron Fluence Evaluation for the Fort Calhoun Unit 1 Reactor Pressure Vessel, July 2000.

(6) Technical Specification 2.3(3)

(7) Article IWB-5000, ASME Boiler and Pressure Vessel Code, Section XI

(8) TR-O-MCM-001, Revision 1, Omaha Public Power District, Fort Calhoun Station Unit No. 1, Evaluation of Irradiated Capsule W-225, August 1980.

(9) TR-O-MCM-002, Omaha Public Power District, Fort Calhoun Station Unit No. 1, Evaluation of Irradiated Capsule W-265, March 1984.

(10) BAW-2226, Omaha Public Power District, Fort Calhoun Station Unit No. 1, Evaluation of Irradiated Capsule W-275, November 1994.

(11) Safety Evaluation of Topical Report CE NPSD-683, Revision 6, "Development of a RCS Pressure and Temperature Limits Report (PTLT) for the Removal of P-T Limits and LTOP Requirements from the Technical Specifications" (TAC No. MA9561).

2.0 LIMITING CONDITIONS FOR OPERATION
2.1 Reactor Coolant System (Continued)
2.1.6 Pressurizer and Main Steam Safety Valves

Applicability
Applies to the status of the pressurizer and main steam safety valves.

Objective
To specify minimum requirements pertaining to the pressurizer and main steam safety valves.

Specifications
To provide adequate overpressure protection for the reactor coolant system and steam system, the following safety valve requirements shall be met:

(1) The reactor shall not be made critical unless the two pressurizer safety valves are operable with their lift settings adjusted to ensure valve opening at 2485 psig ±1% and 2530 psig ±1%.[(1)]

(2) Whenever there is fuel in the reactor, and the reactor vessel head is installed, a minimum of one operable safety valve shall be installed on the pressurizer. However, when in at least the cold shutdown condition, safety valve nozzles may be open to containment atmosphere during performance of safety valve tests or maintenance to satisfy this specification.

(3) At least four of the five Main Steam Safety Valves (MSSVs) associated with each steam generator shall be OPERABLE in MODES 1 and 2. Lift settings shall be at 985 psig +3/-2%, 1000 psig +3/-2%, 1010 psig +3/-2%, 1025 psig +3/-2%, and 1035 psig +3/-2%.[(1)]

a. With less than four of the five MSSVs associated with each steam generator OPERABLE, be in at least HOT STANDBY within 6 hours and HOT SHUTDOWN within an additional 6 hours.

* (4) Two power-operated relief valves (PORVs) shall be operable during heatups and cooldowns when the RCS temperature is less than 515°F, and in Modes 4 and 5 whenever the head is on the reactor vessel and the RCS is not vented through a 0.94 square inch or larger vent, to prevent violation of the pressure-temperature limits designated by Figures 2-1~~A and 2-1B~~.

a. With one PORV inoperable during heatups and cooldowns when the RCS temperature is less than 515°F, restore the inoperable PORV to operable within 7 days or be in cold shutdown within the next 36 hours and depressurize and vent the RCS through at least a 0.94 square inch or larger vent within the following 36 hours.

b. With both PORVs inoperable during heatups and cooldowns when the RCS temperature is less than 515°F, be in cold shutdown within the next 36 hours and depressurize and vent the RCS through at least a 0.94 square inch or larger vent within the following 36 hours.

c. With one PORV inoperable in Modes 4 or 5, within one hour ensure the pressurizer steam space is greater than 53% volume (50.6% or less actual level) and restore the inoperable PORV to operable within 7 days. If adequate steam space cannot be established within one hour, then restore the inoperable PORV to operable within 24 hours. If the PORV cannot be restored in the required time, depressurize and vent the RCS through at least a 0.94 square inch or larger vent within the next 36 hours.

Clean Version of Technical Specifications

TECHNICAL SPECIFICATIONS - FIGURES

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2.0 **LIMITING CONDITIONS FOR OPERATION**
2.1 Reactor Coolant System (Continued)
2.1.1 Operable Components (Continued)

(5) DELETED

(6) Both steam generators shall be filled above the low steam generator water level trip set point and available to remove decay heat whenever the average temperature of the reactor coolant is above 300°F. Each steam generator shall be demonstrated operable by performance of the inservice inspection program specified in Section 3.17 prior to exceeding a reactor coolant temperature of 300°F.

(7) Maximum reactor coolant system hydrostatic test pressure shall be 3125 psia. A maximum of 10 cycles of 3125 psia hydrostatic tests are allowed.

(8) Reactor coolant system leak and hydrostatic test shall be conducted within the limitations of Figure 2-1.

(9) Maximum secondary hydrostatic test pressure shall not exceed 1250 psia. A minimum measured temperature of 73°F is required. Only 10 cycles are permitted.

(10) Maximum steam generator steam side leak test pressure shall not exceed 1000 psia. A minimum measured temperature of 73°F is required.

(11) If no reactor coolant pumps are operating, a non-operating reactor coolant pump shall not be started while T_c is below 385°F unless at least one of the following conditions is met:

2.0 **LIMITING CONDITIONS FOR OPERATION**
2.1 Reactor Coolant System (Continued)

2.1.2 Heatup and Cooldown Rate

Applicability

Applies to the temperature change rates and pressure of the reactor coolant system.

Objective

To specify limiting conditions of the reactor coolant system heatup and cooldown rates.

Specification

The reactor coolant pressure shall be limited during plant operation in accordance with Figure 2-1 and as follows:

(1) Allowable combinations of pressure and temperature (T_c) for a specific heatup rate shall be below and to the right of the applicable limit lines as shown on Figure 2-1.

(2) Allowable combinations of pressure and temperature (T_c) for a specific cooldown rate shall be below and to the right of the applicable limit lines as shown on Figures 2-1.

(3) The heatup rate of the pressurizer shall not exceed l00°F in any one hour period.

(4) The cooldown rate of the pressurizer shall not exceed 200°F in any one hour period.

(5) When any of the above limits are exceeded, the following corrective actions shall be taken:

(a) Immediately initiate action to restore the temperature or pressure to within the limit.

(b) Perform an analysis to determine the effects of the out of limit condition on the fracture toughness properties of the reactor coolant system.

(c) Determine that the reactor coolant system remains acceptable for continued operation or be in cold shutdown within 36 hours.

(6) Before the radiation exposure of the reactor vessel exceeds the exposure for which they apply, Figure 2-1 shall be updated in accordance with the following criteria and procedures:

2.0 **LIMITING CONDITIONS FOR OPERATION**
2.1 Reactor Coolant System (Continued)
2.1.2 Heatup and Cooldown Rate (Continued)

(a) Figure 2-1 is valid for a fast neutron ($E \geq 1MeV$) fluence of $2.15X10^{19}$ n/cm^2 which corresponds to 40 EFPY.

(b) The limit line on the figures shall be updated for a new integrated power period as follows: the total integrated reactor thermal power from startup to the end of the new period shall be converted to an equivalent integrated fast neutron exposure ($E \geq 1$ MeV).

(c) The limit lines in Figure 2-1 shall be moved parallel to the temperature axis (horizontal) in the direction of increasing temperature a distance equivalent to the transition temperature shift during the period since the curves were last constructed. The boltup temperature limit line shall remain at 82°F as it is set by the RT_{NDT} of the reactor vessel flange and not subject to fast neutron flux. The lowest service temperature shall remain at 164°F because components related to this temperature are also not subject to fast neutron flux.

(d) The Technical Specification 2.3(3) shall be revised each time the curves of Figure 2-1 are revised.

Basis

All components in the reactor coolant system are designed to withstand the effects of cyclic loads due to reactor coolant system temperature and pressure changes.(1) These cyclic loads are introduced by normal unit load transients, reactor trips and startup and shutdown operation.

During unit startup and shutdown, the rates of temperature and pressure changes are limited. The design number of cycles for heatup and cooldown is based upon allowable heatup/cooldown rates and cyclic operation.

2.0 LIMITING CONDITIONS FOR OPERATION
2.1 Reactor Coolant System (Continued)
2.1.2 Heatup and Cooldown Rate (Continued)

The maximum allowable reactor coolant system pressure at any temperature is based upon the stress limitations for brittle fracture considerations. These limitations are derived by using the rules contained in Section XI[2] of the ASME Code including Appendix A and G, Westinghouse Electric Company/Combustion Engineering's P-T (W/CE's) limit curve methodology[11], and the rules contained in 10 CFR 50, Appendix G, Fracture Toughness Requirements. This ASME Code assumes that a crack 10-11/16 inches long and 1-25/32 inches deep exists on the inner surface of the vessel. Furthermore, operating limits on pressure and temperature assure that the crack does not grow during heatups and cooldowns.

The reactor vessel beltline material consists of six plates. The nilductility transition temperature (T_{NDT}) of each plate was established by drop weight tests. Charpy tests were then performed to determine at what temperature the plates exhibited 50 ft-lbs. absorbed energy and 35 mils lateral expansion for the longitudinal direction. NRC technical position MTEB-5-2 was used to establish a reference temperature for transverse direction (RT_{NDT}) of -12°F.

The initial RT_{NDT} value for the Fort Calhoun submerged arc vessel weldments was determined to be -56°F with a standard deviation of 17°F. By applying the shift prediction methodology of Regulatory Guide 1.99, Revision 2, a weld material adjusted reference temperature (RT_{NDT}) was established at 10°F based on the mean value plus two standard deviations. The standard deviation was determined by using the root-mean-squares method to combine the margin of 28°F for uncertainty in the shift equation with the margin of 17°F for uncertainty in the initial RT_{NDT} value.

Similar testing was not performed on all remaining material in the reactor coolant system. However, sufficient impact testing was performed to meet appropriate design code requirements[3] and a conservative RT_{NDT} of 50°F has been established.

As a result of fast neutron irradiation in the region of the core, there will be an increase in the T_{NDT} with operation. The techniques used to predict the integrated fast neutron ($E \geq 1$ MeV) fluxes of the reactor vessel are described in Reference 5 with the result that the integrated fast neutron flux ($E \geq 1$ MeV) is 1.73×10^{19} n/cm^2, including tolerance at the inside surface of the critical reactor vessel beltline weld material, over the 40 years design life of the vessel.

Since the neutron spectra and the flux measured at the samples and reactor vessel inside radius should be nearly identical, the measured transition shift for a sample can be applied to the adjacent section of the reactor vessel for later stages in plant life equivalent to the difference in calculated flux magnitude. The maximum exposure of the reactor vessel will be obtained from the measured sample exposure by application of the calibrated azimuthal neutron flux variation. The maximum integrated fast neutron ($E \geq 1$ MeV) exposure of the reactor vessel at the critical reactor vessel beltline location including tolerance is computed to be 1.73×10^{19} n/cm^2 at the vessel inside surface for 40 years operation at

2.0 **LIMITING CONDITIONS FOR OPERATION**
2.1 Reactor Coolant System (Continued)
2.1.2 Heatup and Cooldown Rate (Continued)

1500 MWt and 85% load factor. The predicted shift at this location at the 1/4t depth from the inner surface is projected to be 252°F, including margin, using the shift prediction equation of Regulatory Guide 1.99, Revision 2. The actual shift in T_{NDT} will be re-established periodically during the plant operation by testing of reactor vessel material samples which are irradiated cumulatively by securing them near the inside wall of the reactor vessel as described in Section 4.5.3 and Figure 4.5-1 of the USAR. To compensate for any increase in the T_{NDT} caused by irradiation, limits on the pressure-temperature relationship are periodically changed to stay within the stress limits during heatup and cooldown. Analysis of the three removed irradiated reactor vessel surveillance specimens[(8,9 and 10)], combined with weld chemical composition data and implementation of extreme low radial leakage core loading designs beginning in Cycle 14, indicate that the fluence at the end of 40 Effective Full Power Years (EFPY) at 1500 MWt will be 2.15×10^{19} n/cm^2 on the inside surface of the reactor vessel. This results in a total shift of the RT_{NDT} of 237.76°F, including margin, for the area of greatest sensitivity (weld metal) at the 1/4t location using Regulatory Guide 1.99, Revision 2, and a shift of 187.97°F at the 3/4t location. Operation through fuel Cycle 34 will result in less than 40 EFPY.

The limit lines in Figure 2-1are based on Reference 2, Appendix G, W/CE's methodology for P-T limit curve generation[11], and ASME Code Case N-640 as discussed below.

Reference Stress Intensity Factor

The reference stress intensity factor (K_{IR}) used in the development of the limit lines in Figure 2-1 is based on ASME Code Case N-640. This Code case allows the use of K_{Ic} (lower bound of static initiation critical stress intensity factor) and is an approved exemption by the NRC in accordance with 10CFR50.60(b). K_{Ic} is obtained from a reference fracture toughness curve for reactor pressure vessel low alloy steels as defined in Appendix A to Section XI of the ASME Code and is approximated by the following equation:

$$K_{IC} = 33.20 + 20.734e^{[(0.0200(T-RTNDT)]}$$

Where,

K_{IC} = Crack initiation reference stress intensity factor, Ksi $\sqrt{in}$
T = Temperature at the postulated crack tip, °F
RT_{NDT} = Adjusted reference nil ductility temperature (also called ART) at the postulated crack tip, °F

For any instant during the postulated heatup or cooldown, K_{IC} is calculated using the metal temperature at the tip of the flaw, as well as the value of ART at that flaw location.

2.0 **LIMITING CONDITIONS FOR OPERATION**
2.1 Reactor Coolant System (Continued)
2.1.2 Heatup and Cooldown Rate (Continued)

Regulatory Requirements

The Reference 2, Appendix G equation relating K_{IM}, K_{IT}, and K_{IR} is re-arranged as shown below to solve for the allowable pressure stress intensity factor, K_{IM}, as a function of time with the calculated K_{IR} and K_{IT} values to determine the allowable pressure stress intensity factor and consequently the allowable pressure:

(1) For Service Level A and B operation:

$$K_{IM}=\frac{K_{IR}-K_{IT}}{2}$$

and

(2) For Hydrostatic and Test Conditions when the core is not critical and tests are performed at isothermal conditions (i.e. thermal stress intensity factor, K_{IT}, = 0)

$$K_{IM}=\frac{K_{IR}}{1.5}$$

where,

K_{IM} = Allowable pressure stress intensity factor based on coolant temperature, $Ksi\sqrt{in}$

K_{IR} = Reference stress intensity factor based on coolant temperature, $Ksi\sqrt{in}$

K_{IT} = Thermal stress intensity factor based on coolant temperature, $Ksi\sqrt{in}$

Calculational of P-Allowable

To develop P-T limits, the reactor vessel (RV) beltline region is the only location that receives sufficient neutron fluence to substantially alter the fracture toughness of the RV material. Hence, the beltline region is the most limiting with respect to allowable pressure at any specific temperature. This reduction in fracture toughness is determined using an adjusted reference temperature (ART), which is calculated in accordance with Regulatory Guide 1.99 Revision 2. The allowable pressure is based on the highest ART. The RV beltline region is analyzed assuming a semi-elliptical surface flaw oriented in the axial direction with a depth of one quarter of the RV beltline thickness and an aspect ratio of one to six.

2.0 **LIMITING CONDITIONS FOR OPERATION**
2.1 Reactor Coolant System (Continued)
2.1.2 Heatup and Cooldown Rate (Continued)

This postulated flaw is analyzed at both the inside diameter location (referred to as the 1/4t location) and the outside diameter location (referred to as the 3/4t location) to assure the most limiting condition is achieved. K_{IT} is determined in accordance with Reference 2, Appendix G-2214.3. The thermal stress intensity factors are determined from the calculated temperature profile through the beltline wall using thermal influence coefficients specifically generated for this purpose. The method employed uses a polynomial fit of the temperature profile and superposition using influence coefficients to calculate K_{IT}. The influence coefficients are dependent upon the geometrical parameters associated with the postulated defect, the geometry of the reactor vessel beltline region, and the assumed unit loading. These influence coefficients were calculated using a 2-dimensional finite element model of the reactor vessel. The influence coefficients were corrected for three-dimensional effects using Reference 2, Appendix A procedures. The K_{IT} and K_{IM} are calculated at any time point in a transient using influence coefficients generated by applying unit loads on a finite element model of the reactor vessel beltline region. The influence coefficients are calculations of stress intensity factors at the 1/4t and 3/4t crack depth location under the following unit loads:

a. for K_{IM-P}, pressure load of 1 ksi, $ksi\text{-}\sqrt{in}/ksi$

b. for K_{IT-L}, linear through-wall gradient with peak temperature of 1°F, $ksi\text{-}\sqrt{in}/°F$

c. for K_{IT-Q}, quadratic through-wall gradient with peak temperature of 1°F, $ksi\text{-}\sqrt{in}/°F$

d. for K_{IT-C}, cubic through-wall gradient with peak temperature of 1°F, $ksi\text{-}\sqrt{in}/°F$

Each stress intensity factor is calculated using a standard quarter point element formulation at the respective crack tips. Since all calculations performed are linear, superposition is then used to scale and combine these influence coefficients as necessary to determine the stress intensity factor for a given temperature profile.

2.0 LIMITING CONDITIONS FOR OPERATION
2.1 Reactor Coolant System (Continued)
2.1.2 Heatup and Cooldown Rate (Continued)

In the case of K_{IT}, the through-wall temperature profiles are then fit to the third order polynomial below:

$$T(x) = C_o + C_L (1-x/_h) + C_Q (1-x/_h)2 + C_C (1-x/_h)3$$

where,

$T(x)$ = Temperature at radial location x from inside wall surface, °F
C_o, C_L, C_Q, C_C = Coefficients in polynomial fit, °F
x = Distance through beltline wall, in
h = Beltline wall thickness, in

The coefficients of this polynomial are then combined through the following equation to calculate K_{IT} at the 1/4t and 3/4t locations.

$$K_{IT\text{ - }Total} = C_L * K_{IT\text{-}L} + C_Q * K_{IT\text{-}Q} + C_C * K_{IT\text{-}C}$$

To calculate the allowable pressure, P-Allowable, the resultant $K_{IT\text{-}Total}$ from above in conjunction with Equation (1) from Reference 2, Appendix G-2215, is described as follows.

for Normal Level A and B loads

$$P\text{-}Allowable = \frac{K_{IR} - K_{IT\text{-}Total}}{2 * K_{IM\text{-}P}}$$

for Hydrostatic and Test Conditions, where Isothermal conditions result in$K_{IT\text{-}Total} = 0$

$$P\text{-}Allowable = \frac{K_{IR}}{1.5 * K_{IM\text{-}P}}$$

The P-T limits developed using the method described above account for the temperature differential between the RV base metal and the reactor coolant bulk fluid temperature only. However, uncertainties for instrumentation error, elevation, and flow induced differential pressure differences between the RV beltline and pressurizer are accounted for as follows:

1) Temperature instrumentation uncertainty of 14°F is applied to the entire temperature range of Figure 2-1.

FIGURE 2-2A

DELETED

DELETED

FIGURE 2-2B

DELETED

DELETED

Figure 2-3

This Figure has been deleted.

2.0 **LIMITING CONDITIONS FOR OPERATION**
2.1 Reactor Coolant System (Continued)
2.1.2 Heatup and Cooldown Curves (Continued)

2) Pressure correction factors that account for the difference in pressure between the reactor vessel beltline and pressurizer pressure instrumentation due to elevation and RCP flow are as follows:

   RCS Temperature < 210°F = 61 psi
   RCS Temperature $\geq$ 210°F = 67 psi

3) Below 350°F, pressure instrumentation uncertainty is accounted for in the LTOP system setpoints. Above 350°F, a pressure instrumentation uncertainty of 50 psi is applied to Figure 2-1.

Lowest Service Temperature = 50°F + 100°F + 14°F = 164°F. As indicated previously, an RT_{NDT} for all material with the exception of the reactor vessel beltline was established at 50°F. Reference 2, Section III, NB-2332 requires a lowest service temperature of RT_{NDT} + 100°F for piping, pumps and valves. Below this temperature a pressure of 20 percent of the system hydrostatic test pressure cannot be exceeded. Taking into account pressure correction factors for elevation and flow, this pressure is (.20)(3125) - 61 = 564 psia, where 61 psi is the pressure correction factor.

Boltup Temperature = 10°F + 14°F = 24°F. A conservative value of 82°F will be used and maintained. At pressure below 564 psia, a minimum vessel temperature must be maintained to comply with the manufacturer's specifications for tensioning the vessel head. This temperature is based on RT_{NDT} methods. This temperature corresponds to the measured 10°F RT_{NDT} of the reactor vessel flange, which is not subject to radiation damage, plus 14°F instrument error.

The temperature at which the heatup and cooldown rates change in Figure 2-1 reflects the point at which the most limiting heatup and cooldown rates with respect to the inlet temperature (T_c) change.

2.0 LIMITING CONDITIONS FOR OPERATION
2.1 Reactor Coolant System (Continued)
2.1.2 Heatup and Cooldown Rate (Continued)

References:

(1) USAR, Section 4.2.2

(2) ASME Boiler and Pressure Vessel Code

(3) USAR, Section 4.2.4

(4) USAR, Section 3.4.6

(5) WCAP-15443, Revision 0, Fast Neutron Fluence Evaluation for the Fort Calhoun Unit 1 Reactor Pressure Vessel, July 2000.

(6) Technical Specification 2.3(3)

(7) Article IWB-5000, ASME Boiler and Pressure Vessel Code, Section XI

(8) TR-O-MCM-001, Revision 1, Omaha Public Power District, Fort Calhoun Station Unit No. 1, Evaluation of Irradiated Capsule W-225, August 1980.

(9) TR-O-MCM-002, Omaha Public Power District, Fort Calhoun Station Unit No. 1, Evaluation of Irradiated Capsule W-265, March 1984.

(10) BAW-2226, Omaha Public Power District, Fort Calhoun Station Unit No. 1, Evaluation of Irradiated Capsule W-275, November 1994.

(11) Safety Evaluation of Topical Report CE NPSD-683, Revision 6, "Development of a RCS Pressure and Temperature Limits Report (PTLT) for the Removal of P-T Limits and LTOP Requirements from the Technical Specifications" (TAC No. MA9561).

2.0 **LIMITING CONDITIONS FOR OPERATION**
2.1 Reactor Coolant System (continued)
2.1.6 Pressurizer and Main Steam Safety Valves

Applicability
Applies to the status of the pressurizer and main steam safety valves.

Objective
To specify minimum requirements pertaining to the pressurizer and main steam safety valves.

Specifications
To provide adequate overpressure protection for the reactor coolant system and steam system, the following safety valve requirements shall be met:

(1) The reactor shall not be made critical unless the two pressurizer safety valves are operable with their lift settings adjusted to ensure valve opening at 2485 psig ±1% and 2530 psig ±1%.(1)

(2) Whenever there is fuel in the reactor, and the reactor vessel head is installed, a minimum of one operable safety valve shall be installed on the pressurizer. However, when in at least the cold shutdown condition, safety valve nozzles may be open to containment atmosphere during performance of safety valve tests or maintenance to satisfy this specification.

(3) At least four of the five Main Steam Safety Valves (MSSVs) associated with each steam generator shall be OPERABLE in MODES 1 and 2. Lift settings shall be at 985 psig +3/-2%, 1000 psig +3/-2%, 1010 psig +3/-2%, 1025 psig +3/-2%, and 1035 psig +3/-2%.(1)

a. With less than four of the five MSSVs associated with each steam generator OPERABLE, be in at least HOT STANDBY within 6 hours and HOT SHUTDOWN within an additional 6 hours.

* (4) Two power-operated relief valves (PORVs) shall be operable during heatups and cooldowns when the RCS temperature is less than 515°F, and in Modes 4 and 5 whenever the head is on the reactor vessel and the RCS is not vented through a 0.94 square inch or larger vent, to prevent violation of the pressure-temperature limits designated by Figure 2-1. |

a. With one PORV inoperable during heatups and cooldowns when the RCS temperature is less than 515°F, restore the inoperable PORV to operable within 7 days or be in cold shutdown within the next 36 hours and depressurize and vent the RCS through at least a 0.94 square inch or larger vent within the following 36 hours.

b. With both PORVs inoperable during heatups and cooldowns when the RCS temperature is less than 515°F, be in cold shutdown within the next 36 hours and depressurize and vent the RCS through at least a 0.94 square inch or larger vent within the following 36 hours.

c. With one PORV inoperable in Modes 4 or 5, within one hour ensure the pressurizer steam space is greater than 53% volume (50.6% or less actual level) and restore the inoperable PORV to operable within 7 days. If adequate steam space cannot be established within one hour, then restore the inoperable PORV to operable within 24 hours. If the PORV cannot be restored in the required time, depressurize and vent the RCS through at least a 0.94 square inch or larger vent within the next 36 hours.

*SEE TDB-VIII

References for Attachment 1

Non-proprietary Version

- LTR-PS-01-26, Revision 0, "Evaluation of the Current LTOP Analysis for Revised Technical Specification P-T Limits at the Fort Calhoun Station"
- Report DAR-PS-01-4, Revision 0, "Reactor Coolant System Pressure-Temperature Limits and Low Temperature Overpressure Enable Temperature for 40 Effective Full Power Years for Fort Calhoun Station Unit 1"

LTR-PS-01-26, Revision 0,

"Evaluation of the Current LTOP Analysis for Revised Technical Specification P-T Limits at the Fort Calhoun Station"

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Nuclear Services

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USA

LTR-PS-01-26
Revision 00
November 7, 2001

Mr. Kevin Holthaus
Acting Supervisor, Reactor Performance Analysis
Nuclear Engineering Division
Omaha Public Power District
P. O. Box 399
Fort Calhoun, NE 68023

Subject: Evaluation of the Current LTOP Analysis for Revised Technical Specification P-T Limits at the Fort Calhoun Station

References:
(1) LTR-NEM-01-700, *Proposal for Support Tasks to the Reactor Performance Analysis Group at Fort Calhoun Station*, October 30, 2001
(2) A-FC-PS-0001, Revision 000, OPPD Fort Calhoun Station Unit 1 RCS P-T Limits and LTOP Enable Temperatures for 40 EFPY.
(3) ER-FC-PS-0004, Rev 001 *LTOP Analysis for Fort Calhoun Station for 15 and 20 EFPY: Final Results.*

Dear Mr. Holthaus;

Westinghouse Electric Company (Westinghouse) is pleased to provide the results of the subject evaluation to Omaha Public Power District (OPPD). Specifically, this evaluation has determined acceptable use of a selected set of RCS heatup and cooldown rates for an operating period of 40 EFPY which can replace the current rates in use in the Fort Calhoun Station Technical Specifications. This letter is provided in accordance with the scope of Task 1 in Reference (1).

Figure 1 presents a graphical depiction of RCS pressure-temperature limits for the final selected rates. The objective was to select the best combination of rate limits while remaining bounded by the current LTOP evaluation of record. The rate limits are based upon the results of Reference (2), representing ASME Code approved K_{IC} methodology using projected reactor vessel fluence to 40 EFPY. The decision on the choice of rates was arrived through discussions between Westinghouse, Mr. F. James Jensen of your staff and the Ft. Calhoun Station Operations department. The depiction in Figure 1 is consistent with the request of the Operations department, which presents the allowable rates limits as a smoothed combination of the limiting values for both heatup and

cooldown. Figure 1 is for illustration only; Attachment 1 presents the verified data that can be used by OPPD to develop a graphical presentation of the P-T limits for the Technical Specifications.

Based upon the selected rate limits, Westinghouse performed a formal evaluation of those rates to assure that the current LTOP Evaluation of record, Reference (3), remains valid and conservative. The following rate limits and LTOP criteria were evaluated:

Parameter	Limit
LTOP Enable Temperature :	385°F
Minimum Bolt-up Temperature :	82°F
Maximum Heatup rate:	75°F/hr
Maximum Cooldown rate:	178°F < 100°F/hr
	82°F ≤ 50°F/hr ≤ 178°F

All temperature values are described as indicated cold leg temperature values. Attachment 1 to this letter presents the results of the evaluation of theses criteria. It is concluded that they are bounded by the current LTOP evaluation and can be used to replace the current limits described in the Ft. Calhoun Station Technical Specifications.

This task was performed in accordance with Westinghouse Quality Procedures as Quality Class 1 (Safety-Related). The verification status on the evaluation documented in the Attachment is Complete.

Westinghouse is pleased to support OPPD with this effort and looks forward to future opportunities to assist. Please do not hesitate to contact me at 860-731-6710 if you have questions on the subject.

Sincerely,

Westinghouse Electric Company

F. P. Ferraraccio

Project Manager, Plant Systems

cc: F. J. Jensen, III

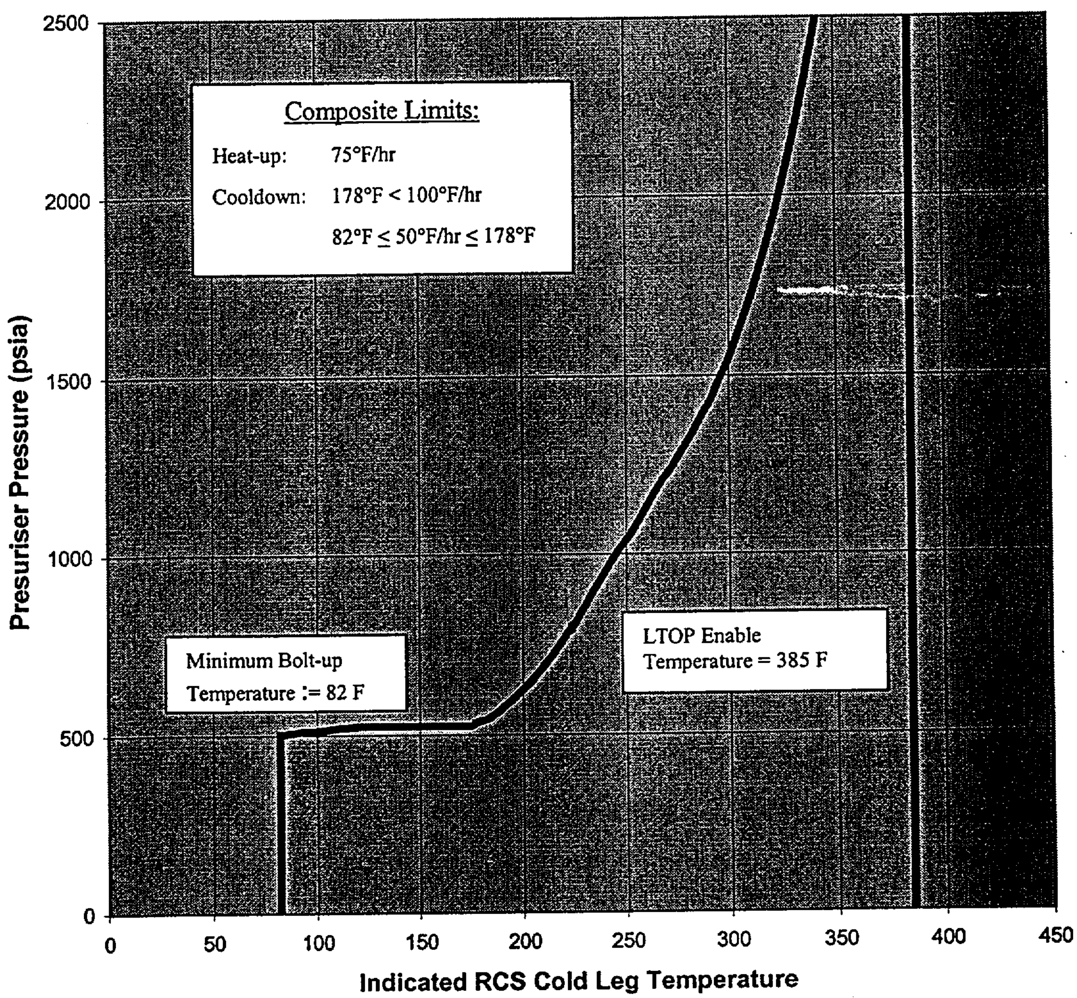
R. O. Doney (W)

C. A. Nielsen (W, RSM)

C. Stuart (W, CM)

Figure 1

Composite RCS P-T Limits for 40 EFPY



Do Not Scale – For Illustration Purposes Only

(Connecting lines between marked data points do not represent analyzed values).

Evaluation of the Current LTOP Analysis with Revised Technical Specification P-T Limits for the Fort Calhoun Station

DESCRIPTION

This evaluation demonstrates that a selected set of revised RCS P-T limits are acceptable for use for the Ft. Calhoun Station Technical Specifications. Acceptability is based upon comparison of the revised set of heatup and cooldown limits with the current limits developed in the LTOP evaluation of record to assure that the current limits are bounding and conservative. Consideration will be given to recognized changes in pressure correction factors and instrumentation uncertainty.

DESIGN INFORMATION

Technical Specification, Section 2.1.2 (the RCS Heatup and Cooldown Rate LCO, which contains the P-T curves), Amendment 199, Reference (A4), describes the current heatup and cooldown rates as:

Maximum Heatup rate:	335°F < 100°F/hr
	82°F ≤ 75°F/hr ≤ 335°F
Maximum Cooldown rate:	285°F < 100°F/hr
	135°F < 30°F/hr ≤ 285°F
	82°F ≤ 10°F/hr ≤ 135°F

Reference (A3) identifies the applicable Technical Specification amendment. The existing LTOP evaluation, documented in the Reference (A1) engineering report, established these rates for use based upon P-T limits for 20 EFPY, using ASME Code K_{IA} methodology. The P-T limit values were developed in Reference (A2). Note that, since the issuance of Reference (A1), OPPD has changed the applicability of the 20 EFPY curves to an extended period of 24.25 EFPY through improved chemistry data of the vessel material and welds, without changing the actual P-T limit values. Note also that OPPD has not invoked the ASME Code Case 514 in the implementation of the current or proposed Technical Specification P-T limits

DESIGN INPUTS

The body of this document identifies the selection of new heatup and cooldown rates for evaluation. These are based upon the ASME Code K_{IC} methodology and are developed in Reference (A5).

Current RCS temperature measurement uncertainty is a value of 14°F, documented in Reference (A3). The temperature measurement uncertainty value used in the current LTOP evaluation is 16°F, in Reference (A1).

APPROACH

The proposed K_{IC} P-T Limit heatup and cooldown rate data are extracted from Reference (A5) and listed in Table 1. Using linear interpolation, the corresponding pressure value for the 100°F/hr cooldown at 178°F is determined as 525 psia. For practical purposes, a horizontal pressure line at 525 psia is used to connect the 100°F/hr and 50°F/hr limits. Using linear interpolation, the corresponding temperature value on the 50°F/hr limit is 118°F. A constant pressure (horizontal) limit in the range between 118°F and 178°F is bounding of the 50°F/hr limit.

Comparison of each heatup rate value with the 100°F/hr cooldown rate values at 244°F and greater shows that the heatup rate pressure values at corresponding temperature values are bounding of the 100°F/hr cooldown rate values.

Thus, based upon the above, a complete composite curve of the selected K_{IC} heatup and cooldown limits is summarized in the last column of Table 1.

Table 2 presents a summary of the Technical Specification K_{IA} heatup and cooldown data from Reference A2. Separate composite heatup and cooldown limit curves are presented, as well.

In Table 3, the three columns of composite data are restated to facilitate comparison between current and proposed limits.

It is noted that the K_{IA} and K_{IC} temperature values differ by 2 degrees. This is a consequence of the change in temperature measurement uncertainty. Because the current temperature uncertainty value is less than the older value used in the LTOP analysis of record, direct comparison of pressure values at near corresponding temperature values is conservative, relative to finding the K_{IA} limit bounding.

The data presented in Tables 1, 2 and 3 as extracted from their respective references have been adjusted for pressure correction to the pressurizer. Consistent with current OPPD practice, the pressure correction factors do not contain pressurizer pressure instrument uncertainty for values less than the LTOP enable temperature. Note, however, that the last data point in Tables 1 and 3 (354°F) *does* use a correction fact that contains pressurizer pressure instrument uncertainty because of the revised (lower) value for LTOP enable temperature (350°F) established in Reference (A5). Since the values are conservative as is for the purposes of this evaluation, they were used without adjusting the correction factor to remove the pressurizer pressure instrument uncertainty.

RESULTS

The data of Table 3 shows that at all temperature values, the K_{IC} composite curve is bounded by either one or both the K_{IA} heatup or cooldown composite curves.

The LTOP Enable temperature and Bolt-up temperature values of 385 °F and 82 °F respectively represent minimum indicated values relative to the LTOP Evaluation. Since the current value for RCS Temperature instrument uncertainty is less than that used in the LTOP Evaluation, the current values are conservative. (by at least 2 °F)

CONCLUSION

It is concluded that the selected 40 EFPY K_{IC} heatup and cooldown rate selection:

Maximum Heatup rate:	75°F/hr
Maximum Cooldown rate:	178°F < 100°F/hr
	82°F ≤ 50°F/hr ≤ 178°F

is an acceptable alternative to the current 20 EFPY K_{IA} based limits in use in the current Technical Specifications.

It is also concluded that the current values for Bolt-up temperature and LTOP Enable temperature:

Parameter	Limit
LTOP Enable Temperature :	385°F
Minimum Bolt-up Temperature :	82°F

are applicable in consideration of the change in RCS temperature instrumentation uncertainty.

QUALITY CLASS

Attachment 1 has been reviewed and approved for use as safety related material in accordance with Westinghouse quality assurance procedures for Other Design Documents. Appropriate reviewer checklists are contained in Attachment 2 to this letter. Attachment 2 must be maintained with the QA Record copy but are not necessary for general distribution.

VERIFICATION STATUS: COMPLETE

Verified: James E. Robertson James E. Robertson Date: 11/08/01

REFERENCES

A1. ER-FC-PS-0004, Rev 001, *LTOP Analysis for Fort Calhoun Station for 15 and 20 EFPY: Final Results, March 2001.*

A2. O-MPS-90-043, *P/T Limits and Pressure Transient Analysis for Fort Calhoun Station – Final Reports*, A. Ostrov to K. Holthaus, June 14, 1990.

A3. NED-DEN-01-0147, *Requested Data to be incorporated into the 40 EFPY Pressure and Temperature (P-T) Limit Curve Calculation*, K. C. Holthaus to F. P. Ferraraccio, October 26, 2001

A4. Fort Calhoun Station Unit 1 Technical Specifications, Amendment 199, Validated by Reference A3 as current amendment.

A5. A-FC-PS-0001, Revision 000, *OPPD Fort Calhoun Station Unit 1 RCS P-T Limits and LTOP Enable Temperatures for 40 EFPY.*

Table 1

K_{IC} Heatup and Cooldown Data (P allowable, corrected)					
RCS Temp (Corrected)	75 °F/hr Heatup	100 °F/hr Cooldown	50 °F/hr Cooldown		Composite Curve
(°F)	(ksi)	(ksi)	(ksi)		(ksi)
64	0.638	0.360	0.498		0.498
74	0.640	0.364	0.502		0.502
84	0.641	0.369	0.505		0.505
94	0.609	0.376	0.510		0.510
104	0.592	0.383	0.515		0.515
114	0.588	0.392	0.522		0.522
118*			0.525		0.525
124	0.588	0.404	0.530		0.525
134	0.591	0.417	0.540		0.525
144	0.599	0.434	0.553		0.525
154	0.610	0.455	0.568		0.525
164	0.626	0.480	0.586		0.525
174	0.646	0.510	0.609		0.525
178*		0.525			0.525
184	0.671	0.548	0.636		0.548
194	0.703	0.593	0.669		0.593
204	0.743	0.649	0.710		0.649
214	0.791	0.717	0.760		0.717
223.9	0.850	0.799	0.821		0.799
224	0.845	0.794	0.815		0.794
234	0.918	0.895	0.890		0.895
244	0.990	0.990	0.981		0.990
254	1.072	1.072	1.072		1.072
264	1.172	1.172	1.172		1.172
274	1.259	1.293	1.293		1.259
284	1.355	1.442	1.442		1.355
294	1.471	1.623	1.623		1.471
304	1.613	1.845	1.845		1.613
314	1.786	2.116	2.116		1.786
324	1.998	2.447	2.447		1.998
334	2.258	2.851	2.851		2.258
344	2.575	3.344	3.344		2.575
354	2.918	3.867	4.007		2.918

* - Represents an interpolated value.

Table 2

K_{IA} Heatup and Cooldown Data
(P allowable, corrected)

RCS Temp (Corrected)	75 °F/hr Heatup	100 °F/hr Heatup	Composite Heatup Curve	10 °F/hr Cooldown	30 °F/hr Cooldown	100 °F/hr Cooldown	Composite Cooldown Curve
(°F)	(ksi)	(ksi)	(ksi)	(ksi)	(ksi)	(ksi)	(ksi)
66	0.5160		0.5160	0.4881			0.4881
76	0.5185		0.5185	0.4907			0.4907
86	0.5214		0.5214	0.4937			0.4937
96	0.5247		0.5247	0.4972			0.4972
106	0.5285		0.5285	0.5012			0.5012
116	0.5112		0.5112	0.5058			0.5058
126	0.4963		0.4963	0.5112	0.4577		0.5112
* 135				* 0.5168	* 0.4637		* 0.4637
136	0.4875		0.4875	0.5174	0.4644		0.4644
146	0.4844	0.4586	0.4844		0.4723		0.4723
156	0.4855	0.4529	0.4855		0.4813		0.4813
166	0.4907	0.4512	0.4907		0.4918		0.4918
176	0.4996	0.4533	0.4996		0.5039		0.5039
186	0.5118	0.4589	0.5118		0.5179		0.5179
196	0.5281	0.4686	0.5281		0.5341		0.5341
206	0.5478	0.4818	0.5478		0.5527		0.5527
216	0.5717	0.4987	0.5717		0.5744		0.5744
226	0.5942	0.5137	0.5942		0.5933		0.5933
236	0.6274	0.5389	0.6274		0.6222		0.6222
246	0.6668	0.5697	0.6668		0.6556		0.6556
256	0.7122	0.6060	0.7122		0.6942		0.6942
266	0.7651	0.6483	0.7651		0.7388		0.7388
276	0.8267	0.6977	0.8267		0.7904	0.7316	0.7904
* 285					* 0.8441	* 0.8054	* 0.8054
286	0.8804	0.7549	0.8804		0.8501	0.8136	0.8136
296	0.9407	0.8225	0.9407			0.908	0.908
306	1.0104	0.9007	1.0104			1.0104	1.0104
316	1.0909	0.9910	1.0909			1.0909	1.0909
326	1.1840	1.0952	1.1840			1.184	1.184
* 335	* 1.2693	* 1.2033	* 1.2033				
336	1.2788	1.2153	1.2153			1.2917	1.2917
346		1.3565	1.3565			1.4162	1.4162
356		1.4626	1.4626			1.56	1.56
366		1.5692	1.5692			1.7263	1.7263
376		1.6918	1.6918			1.9186	1.9186
386		1.8329	1.8329			2.1409	2.1409
396		1.9986	1.9986			2.3979	2.3979
406		2.1895	2.1895			2.6949	2.6949

* - Represents an interpolated value.

Table 3

K_{IC} Heatup and Cooldown Data (P allowable, corrected)	
RCS Temp (Corrected)	Composite Curve
(°F)	(ksi)
64	0.498
74	0.502
84	0.505
94	0.510
104	0.515
114	0.522
118*	0.525
124	0.525
134	0.525
144	0.525
154	0.525
164	0.525
174	0.525
178*	0.525
184	0.548
194	0.593
204	0.649
214	0.717
223.9	0.799
224	0.794
234	0.895
244	0.990
254	1.072
264	1.172
274	1.259
284	1.355
294	1.471
304	1.613
314	1.786
324	1.998
334	2.258
344	2.575
354	2.918

K_{IA} Heatup and Cooldown Data (P allowable, corrected)		
RCS Temp (Corrected)	Composite Heatup Curve	Composite Cooldown Curve
(°F)	(ksi)	(ksi)
66	0.5160	0.4881
76	0.5185	0.4907
86	0.5214	0.4937
96	0.5247	0.4972
106	0.5285	0.5012
116	0.5112	0.5058
126	0.4963	0.5112
* 135		* 0.4637
136	0.4875	0.4644
146	0.4844	0.4723
156	0.4855	0.4813
166	0.4907	0.4918
176	0.4996	0.5039
186	0.5118	0.5179
196	0.5281	0.5341
206	0.5478	0.5527
216	0.5717	0.5744
226	0.5942	0.5933
236	0.6274	0.6222
246	0.6668	0.6556
256	0.7122	0.6942
266	0.7651	0.7388
276	0.8267	0.7904
* 285		* 0.8054
286	0.8804	0.8136
296	0.9407	0.9080
306	1.0104	1.0104
316	1.0909	1.0909
326	1.1840	1.1840
* 335	* 1.2033	
336	1.2153	1.2917
346	1.3565	1.4162
356	1.4626	1.5600

* - Represents an interpolated value.

Report DAR-PS-01-4, Revision 0,

"Reactor Coolant System Pressure-Temperature Limits
and
Low Temperature Overpressure Enable Temperature
for
40 Effective Full Power Years
for
Fort Calhoun Station Unit 1

REPORT DAR-PS-01-4, Rev. 0

Reactor Coolant System Pressure–Temperature Limits and Low Temperature Overpressure Enable Temperature for 40 Effective Full Power Years for Fort Calhoun Station Unit 1

PREPARED FOR:

Omaha Public Power District

PREPARED BY:

Westinghouse Electric Company, LLC

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Westinghouse Non-Proprietary Class 3

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1.0 Introduction

This report establishes:

- Reactor Coolant System (RCS) pressure-temperature limits for preventing brittle fracture
- Low Temperature Overpressure Protection (LTOP) enable temperatures.

for the Ft. Calhoun Station (FCS) of Omaha Public Power District (OPPD).

The analysis that establishes pressure-temperature (P-T) limits was performed in accordance with 10 CFR 50 Appendix G (Reference 1) and is based upon the principles of Linear Elastic Fracture Mechanics found in the ASME Boiler and Pressure Vessel Code Section XI, Appendix G. Note that in 1995, 10 CFR 50 redirected compliance with ASME Code Section III, Appendix G (Reference 2) to ASME Code Section XI, Appendix G (Reference 3). It should also be noted that the P-T limit analysis used the crack initiation reference stress intensity factor, K_{IC}, for determining the fracture toughness of the material. The use of K_{IC} as the basis for establishing the reference fracture toughness limit, K_{IR}, value for the reactor vessel is outlined in ASME Code Case N-640 (Reference 4).

The analysis that establishes LTOP enable temperatures uses ASME Code Section XI, Appendix G as a guide.

2.0 Summary of Results

P-T limits for Fort Calhoun Station Unit 1 were obtained for the RCS for 40 effective full power years (EFPY).

The beltline region was analyzed with the PTCURVE computer code and the following P-T limits were obtained with pressure and temperature correction factors applied:

1) Beltline P-T limits for an isothermal condition.
2) Beltline P-T limits for heatup rates of 10°F/hr, 20°F/hr, 30°F/hr, 40°F/hr, 50°F/hr, 75°F/hr and 100°F/hr.
3) Beltline P-T limits for cooldown rates of 10°F/hr, 20°F/hr, 30°F/hr, 40°F/hr, 50°F/hr, 75°F/hr, and 100°F/hr.
4) Beltline P-T limits for hydrostatic testing.

The above results are presented in Figures 1 through 3 and the numerical values for the figures are presented in Tables 1 through 3.

In addition, results for the beltline region are presented in Figures 4 through 6 without correction factors. Numerical values for the figures are presented in Tables 4 through 6.

Figures 7 and 8, respectively, present the composite P-T limit heatup and cooldown curves. These figures use the results from items 1) through 4) above and present the minimum boltup temperature, minimum pressure, lowest service temperature, and flange limits as determined herein. The figures include pressure and temperature correction factors. Core critical limits are identified in Section 4.4, but are not shown on Figures 7 and 8.

Reference 5 requested that a bounding boltup temperature of 50°F be used. Since analyses demonstrated acceptability of this temperature, 50°F is used to specify the minimum boltup temperature (64°F with temperature correction applied) shown on Figures 7 and 8..

LTOP enable temperatures were determined and are documented in Section 4.9 of this report. Reference 5, however, requested use of a bounding LTOP enable temperature of 350°F. Analysis results presented in Section 4.9 demonstrate that a bounding LTOP enable temperature of 350°F is acceptable.

3.0 Basic Data and Assumptions

3.1 Reactor Vessel Data

Design Pressure = 2500 psia
Design Temperature = 650°F
Operating Pressure = 2100 psia
Beltline Thickness = 7.1250 inch
Inside Radius (w/o clad) = 70.8567 inch
Cladding Thickness = 7/32 inch

3.2 Material – SA533 Grade B Class 1

Thermal Conductivity = 23.8 Btu/hr-ft-°F
Young Modulus = 28.0 x E6 psi
Coefficient of Thermal Expansion = 7.77 x E-6 in/in-°F
Yield Stress = 44,500 psi

3.3 Temperature Correction Factor

Reference 5 provided an RCS temperature instrumentation indication uncertainty of dT = 14.0°F.

3.4 Pressure Correction Factors

The pressure correction factors are based on:

1) static head due to the elevation differences between the reactor vessel and pressurizer,
2) hydraulic pressure drop between the reactor vessel and hot leg surge line nozzle due to RCP flow,
3) flow rate uncertainty, and
4) in some cases pressure instrument uncertainties (results are generated with and without pressure instrument uncertainty included).

Static head is determined as the elevation difference from the inside bottom of reactor vessel to the pressurizer upper instrument nozzle. For the Fort Calhoun Station, this value is 762.5 inches.

The Fort Calhoun Station minimum bolt-up temperature is 50 °F, by Reference 5. Specific volume at this minimum temperature is υ= 0.01602 ft^3/lbm. Thus:

$$\Delta P_{elev} = 762.5 \text{ in} / [12 \text{ in/ft}] / [0.01602 \text{ ft}^3\text{/lbm}] / [144 \text{ in}^2\text{/ft}^2] = 27.54 \text{ psid}$$

The RCP flow induced pressure drops from the reactor vessel inlet to the point where the surge line meets the hot leg were calculated in a separate document accounting for Fort Calhoun Station. The pressure drops are available for the following RCP operating configurations assuming the most limiting combinations.

Number of RCPs Operating	RCS Temperature Range	Pressure Drop
2 RCP	$T_c < 210$ °F	28.12 psid
3 RCP	$T_c \geq 210$ °F:	34.19 psid

These pressure drop values are associated with RCS conditions following removal of the steam generator orifice plates. A pressure uncertainty of 5 psi due to flow uncertainty is included in each of the pressure correction factors.

Consistent with past practice at Fort Calhoun Station, the P-T limits are calculated without accounting for pressure indication loop uncertainties for the portion of the P-T limits below the LTOP enable temperature, and will include pressure indication uncertainty at temperature equal and above. Since the LTOP enable temperature is greater than 210 °F, the factor for 3 operating RCPs will address both cases with and without pressure instrument uncertainty.

Overall uncertainty for a combined flow uncertainty and pressure instrumentation uncertainty is determined by:

$$\Delta P_{unc} = ((\Delta P_{flow\ unc})^2 + (\Delta P_{instr\ unc})^2)^{1/2}$$

The updated pressurizer pressure instrument uncertainty, $\Delta P_{instr\ unc}$, is provided by Reference 5 as a single value of 50 psid applicable to either wide or narrow range channels.

$$\Delta P_{unc} = ((5\ psid)^2 + (50\ psid)^2)^{1/2} = 50.25$$

The following pressure correction factors are to be subtracted from the reactor vessel Appendix G limits to obtain the RCS P-T limitations in terms of pressurizer pressure. These factors are calculated using the following equation:

$$\Delta P_{corr} = \Delta P_{elev} + \Delta P_{flow} + \Delta P_{unc}$$

(a) Without pressurizer pressure instrument uncertainty (due to flow and elevation only)
2 RCPs Operating, $T_c < 210$ °F,
ΔP_{corr} = 27.54 psid + 28.12 psid + 5 psid = 60.7 psid (use 61 psid)

3 RCPs Operating, $T_c \geq 210$ °F,
ΔP_{corr} = 27.54 psid + 34.19 psid + 5 psid = 66.7 psid (use 67 psid)

(b) With pressurizer pressure instrument uncertainty
3 RCPs Operating, $T_C \geq 210$ °F,
ΔP_{corr} = 27.54 psid + 34.19 psid + 50.25 psid = 112.0 psid (use 112 psid)

Note that pressure correction factors are not applied above the pressure of 3,500 psi in the tabular data of Tables 1, 2 and 3 due to the practical limits of plant operation.

3.5 Adjusted RT_{NDT} Values

Adjusted RT_{NDT} Values (Reference 5) are:

EFPY	ADJ RT_{NDT} (°F)	
	1/4t	3/4t
40	237.76	187.97

3.6 Film Coefficient for Heat Transfer

The film coefficient for heat transfer equals 1000 Btu/hr-ft^2-°F.

3.7 Steel Density

The density of typical steel is 490 lbm/ft^3.

3.8 Steel Specific Heat

The specific heat of typical steel is 0.122 Btu/lb-°F.

4.0 Analyses

4.1 Description of Analysis Method

The PTCURVE computer code is used to calculate the reactor vessel beltline pressure-temperature limits and is described below.

The analytical procedure for developing reactor vessel P-T limits uses the methods of Linear Elastic Fracture Mechanics (LEFM) and the guidance found in the ASME Boiler and Pressure Vessel Code Section XI, Appendix G (Reference 3) in accordance with the requirements of 10 CFR Part 50 Appendix G (Reference 1). For these analyses, the Mode I (opening mode, according to fracture mechanics terminology) stress intensity factors are used for the solution basis.

Final RCS P-T limits are established based upon the beltline P-T limits (i.e. with no correction factors) and are adjusted to pressurizer pressure conditions. The adjustment accounts for temperature instrument uncertainties and the pressure effects of RCS flow-related hydraulic pressure drop and the pressurizer-to-beltline region elevation difference. Pressurizer pressure instrumentation error is addressed in the correction factors only at temperatures greater than the LTOP enable temperature.

The reactor vessel beltline region is analyzed assuming a semi-elliptical surface flaw oriented in the axial direction with a depth of one quarter of the reactor vessel beltline thickness. The assumed flaw has an aspect ratio of one to six. The postulated flaw is analyzed at both the inside diameter location (referred to as the 1/4 t location) and the outside diameter location (referred to as the 3/4 t location) to assure the most limiting condition is evaluated.

At each of the postulated flaw locations, the Mode I stress intensity factor, K_I, produced by each of the specified loadings is calculated and the summation of the K_I values is compared to a reference stress intensity, K_{IR}, which is the critical value of K_I for the material and temperature involved. This method produces a relationship of pressure versus temperature for reactor vessel operating limits which conservatively preclude brittle fracture.

K_{IR} is obtained from a reference fracture toughness curve for reactor vessel low alloy steels and is defined in Appendices A and G to Section XI of the ASME Code (References 6 and 3, respectively). K_{IR} is determined by two properties, K_{IA} and $K_{IC,}$ which represent critical values of the stress intensity factor. In this calculation, K_{IR} is defined as $K_{IC,}$ which is defined as the lower bound of static initiation critical K_I values measured as a function of temperature.

This governing curve is defined by the following expression:

$$K_{IR} = K_{IC} = 33.20 + 2.806e^{[.0200\,(T - RT_{NDT} + 100)]}$$

where

K_{IR}	=	reference stress intensity factory, Ksi $\sqrt{in}$
K_{IC}	=	crack initiation reference stress intensity factory, Ksi $\sqrt{in}$
T	=	temperature at the assumed flaw tip, °F
RT_{NDT}	=	adjusted reference nil ductility temperature at assumed flaw tip, °F

For any instant during the postulated heatup or cooldown, K_I is calculated at the metal temperature and at the adjusted RT_{NDT} at the tip of the flaw. The temperature distribution and the temperature at the flaw tip are calculated using a one dimensional three noded isoparametric finite element suitable for one dimensional radial conduction-convection heat transfer analysis.

The fracture mechanics algorithms make use of a superposition technique using influence coefficients for calculating the Mode I stress intensity factors. In general, the thermal stress intensity factors are found using the temperature difference through the wall as a function of transient time. They are then subtracted from the available K_{IR} value to find the allowable pressure stress intensity factor and consequently the allowable pressure.

In general, the expression used to derive P-T limits is:

$$2K_{IM} + K_{IT} < K_{IR} \quad \text{(Reference 3)}$$

where,

K_{IM}	=	Allowable Pressure Stress Intensity, Ksi $\sqrt{in}$
K_{IT}	=	Thermal Stress Intensity, Ksi $\sqrt{in}$

K_{IR} = Reference Stress Intensity, Ksi $\sqrt{in}$

The superposition technique used here is temperature profile based rather than stress profile based which is typically used. A third order polynomial fit to the temperature distributions in the wall was used and is given by:

$$T(x) = C_0 + C_1(1-\frac{x}{h}) + C_2(1-\frac{x}{h})^2 + C_3(1-\frac{x}{h})^3$$

where,

$T(x)$ = Temperature at radial location x from inside wall surface

C_0-C_3 = Coefficients in polynomial fit

x = Distance through beltline wall, in

h = Beltline wall thickness, in

The unit K_I values are calculated for each term of the polynomial using a two dimensional finite element code. These unit values are used to determine the total K_I value for the applied loads under any general temperature profile in the wall that occurs during the thermal transient.

The thermal stress intensity factor is represented by the following expression:

$$K_{IT}(a) = \sum_{i=0}^{3} C_i K_i^* \sqrt{\pi a}$$

where,

K_{IT} = thermal stress intensity factor

C_i = coefficients in polynomial fit

K_i^* = polynomial influence coefficients

a = crack depth

Temperature based influence coefficients for determination of the thermal stress intensity factor, K_{IT}, are used. These were computed using a two dimensional reactor vessel model with a crack adjusted to account for three dimensional effects using methods from Reference 7.

Isothermal and transient conditions were analyzed at the selected values of 40 EFPY. The cooldown transients, which were analyzed at rates of 10°F/hr, 20°F/hr, 30°F/hr, 40°F/hr, 50°F/hr, 75° /hr and 100°F/hr, begin at a bulk coolant temperature of 550°F and terminate at 50°F. The heatup transients, which were analyzed at rates of 10°F/hr, 20°F/hr, 30°F/hr, 40°F/hr, 50° /hr, 75°F/hr, and 100°F/hr, begin at a bulk temperature of 50° and terminate at 550°F. The hydrostatic limits were obtained only for the isothermal condition.

4.2 Heatup Limit Analysis

During heatup, the thermal bending stress is compressive at the reactor vessel inside wall and is tensile at the reactor vessel outside wall. Internal pressure creates a tensile stress at the inside wall and outside wall locations. Consequently, the outside wall location has the larger total stress when compared to the inside wall. However, neutron embrittlement, shift in material RT_{NDT}, and reduction in fracture toughness are greater at the inside location than the outside. Therefore, results from both the inside and outside flaw locations must be compared to assure that the most limiting condition is recognized.

As described in the cooldown case, the reference stress intensity is calculated at the metal temperature and the adjusted RT_{NDT} at the tip of the flaw. Using a finite element method for the heat transfer analysis, the temperature profile through the wall and the metal temperatures at the tip of the flaw are calculated for the transient history. This information is used to calculate the thermal stress intensity factor at the 1/4t and 3/4t locations using the calculated wall gradient and thermal influence coefficients. The allowable pressure stress intensity is then determined by superposition of the thermal stress intensity factor with the available reference stress intensity at the flaw tip. The allowable pressure is derived from the calculated allowable pressure stress intensity factor.

It is interesting to note that a sign change occurs in the thermal stress through the reactor vessel beltline wall. Assuming a reference flaw at the 1/4 t location, the thermal stress tends to alleviate the pressure stress indicating the isothermal steady state condition would represent the limiting P-T limit. However, the isothermal condition may not always provide the limiting pressure-temperature limit for the 1/4 t location during a heatup transient. This is due to the difference between the base metal temperature and the Reactor Coolant System (RCS) fluid temperature at the inside wall. For a given heatup rate (non-isothermal), the differential temperature through the clad and film increases as a function of thermal rate, resulting in a crack tip temperature which is lower than the RCS fluid temperature. Therefore to ensure the accurate representation of the 1/4t pressure-temperature limit during heatup, both the isothermal and heatup rate dependent pressure-temperature limits are calculated to ensure the limiting condition was

recognized. These limits account for clad and film differential temperatures and for the gradual buildup of wall differential temperatures with time, as do the cooldown limits.

To develop minimum pressure-temperature limits for the heatup transient, the isothermal conditions at 1/4t and 3/4t, 1/4 t heatup, and 3/4 t heatup pressure-temperature limits are compared for a given thermal rate. Then the most restrictive pressure-temperature limits are combined resulting in a minimum limit curve for the reactor vessel beltline for the heatup event.

Table 1 provides the results for isothermal and 10°F/hr through 100°F/hr heatup pressure-temperature limits. The table provides the allowable pressure based on the inside and outside flaw locations versus reactor coolant temperature. These values have temperature and pressure correction factors applied. The allowable pressure has units of ksi while temperature has units of degrees F. Note that Table 1 data applies the pressure correction factor with pressurizer pressure instrument uncertainty (Section 3.4 (b)) at pressure values greater than the LTOP Enable Temperature of 350°F (Reference 5). Table 4 presents the P-T results for conditions at the beltline with no correction factors applied.

4.3 Cooldown Limit Analysis

During cooldown, membrane and thermal bending stresses act together in tension at the reactor vessel inside wall. This results in the pressure stress intensity factor, K_{IM}, and the thermal stress intensity factor, K_{IT}, acting in unison creating a high stress intensity. At the reactor vessel outside wall, the tensile pressure stress and the compressive thermal stress act in opposition, resulting in a lower total stress than at the inside wall location. Also, neutron embrittlement, the shift in RT_{NDT} and the reduction in fracture toughness are less severe at the outside wall compared to the inside wall. Consequently, the inside flaw location is more limiting for the cooldown event.

Using the material metal temperature and adjusted RT_{NDT} at the 1/4t and 3/4t locations, the reference stress intensities are determined. From the finite element method used for the heat transfer analysis, the through wall temperature gradient is calculated for the assumed cooldown rate to determine the thermal stress intensity factor. In general, the thermal stress intensity factors are found using the temperature difference through the wall as a function of transient time. They are then subtracted from the available K_{IR} value to find the allowable pressure stress intensity factor and, consequently, the allowable pressure.

The cooldown pressure-temperature curves are thus generated by calculating the allowable pressure on the reference flaw at the 1/4t and 3/4t locations based upon,

$$K_{IM} = \frac{K_{IR} - K_{IT}}{2}$$

where,

K_{IM} = Allowable pressure stress intensity as a function of coolant temperature, Ksi $\sqrt{in}$

K_{IR} = Reference stress intensity as a function of coolant temperature, Ksi $\sqrt{in}$

K_{IT} = Thermal stress intensity as a function of coolant temperature, Ksi $\sqrt{in}$

The isothermal P-T limit must be calculated to develop a minimum P-T limit for the cooldown event. The isothermal P-T limit is then compared to the P-T limit associated with a cooling rate and the more restrictive allowable P-T limit is chosen, resulting in a minimum limit curve for the reactor vessel beltline.

Table 2 provides the results for isothermal and 10°F/hr through 100°F/hr cooldown P-T limits. The table provides the allowable pressure based upon the inside and outside flaw locations versus reactor coolant temperature. These values have pressure and temperature correction factors applied. The allowable pressure has units of ksi while temperature has units of degrees F. Note that Table 2 data applies the pressure correction factor with pressurizer pressure instrument uncertainty at pressure values greater than the LTOP Enable Temperature of 350°F (Reference 5). Table 5 presents the P-T results for conditions at the beltline with no correction factors applied.

4.4 Hydrostatic Test Limits and Core Critical Limits

Hydrostatic test limits have been calculated for 40 EFPY using the methodology of the ASME Boiler and Pressure Vessel Code, Section XI, Appendix G (Reference 3). The governing equation for determining hydrostatic test limits is:

$$1.5K_{IM} + K_{IT} < K_{IR}$$

The procedure is the same as when calculating normal operation heatup and cooldown limits with the exception of the factor of safety applied to the allowable pressure stress intensity (K_{IM}). The PTCURVE code was used for this calculation by changing the applied factor of safety from 2.0 for normal operation to 1.5 for hydrostatic limits.

The purpose of the hydrostatic test limit is to establish the minimum temperature required at the corresponding hydrostatic test pressure. It is recommended practice that the inservice hydrostatic test for CE NSSS designs be performed at a test pressure corresponding to 1.1 times the operating pressure with the reactor core not critical.

Under these conditions, 10CFR50, Appendix G requires that the minimum temperature for the reactor vessel (RV) must be at least as high as the RT_{NDT} for the limiting material in the closure flange region plus 90 °F. However, the beltline hydrostatic test limits at the recommended test pressure are more limiting. Hence, it is only necessary to show the beltline inservice hydrostatic test limits in the vicinity of this pressure.

To define minimum temperature criteria for core critical operation, Appendix G to 10 CFR Part 50 specifies the following P-T limits. In the case when RCS pressure is less than or equal to 20% of the preservice hydrostatic test pressure (PHTP), the minimum temperature requirement for the RV must be at least as high as the RT_{NDT} for the limiting material in the stressed region of the closure flange by bolt preload plus 40°F, or the minimum permissible temperature for the inservice hydrostatic pressure test, whichever is larger. In the case when the RCS pressure is greater than 20% of the PHTP, the minimum temperature requirement for the RV must be at least as high as the RT_{NDT} for the limiting material in the stressed region of the closure flange by bolt preload plus 160°F, or the minimum permissible temperature for the inservice hydrostatic pressure test, whichever is larger.

According to 10 CFR Part 50, Appendix G the calculation below specifies P-T limits for core critical operation to provide additional margin during actual power operation.

Inservice hydrostatic pressure
= (1.1 x Operating Pressure) + instrumentation uncertainty
= (1.1 x 2100 psia) + 50 psi = 2360 psia

Note that the pressure instrumentation uncertainty is 50 psi from Reference5.

The minimum temperature for core critical operation and hydrostatic test is the temperature corresponding to the inservice hydrostatic pressure which is 300°F. This temperature was obtained from Table 3 by interpolating the temperature values to the pressure of 2,360 psi, and is corrected for instrument uncertainty of 14 °F (Reference 5).

Note, that the core critical limits established herein are based soley upon fracture mechanics considerations, and do not consider core physics safety analyses which can control the temperature at which the core can be brought critical.

Hydrostatic test limits are tabulated in Table 3 and are corrected with the temperature and pressure correction factors from Sections 3.3 and 3.4, respectively. These values are presented graphically in Figure 3.

4.5 Minimum Boltup Temperature

The minimum boltup temperature, from Reference 3, should be at least the initial RT_{NDT} temperature for the material in the stressed (flange) region plus any effects of irradiation. This is for pressures less than twenty percent of pre-service hydrostatic pressure (625 psia uncorrected). Since there is no meaningful irradiation effect in the flange region, the minimum boltup temperature with instrument uncertainty is:

$$\text{Minimum Boltup Temperature} = \text{Initial } RT_{NDT} + dT$$

$$= 10°F + 14°F = 24°F$$

The initial RT_{NDT} of 10°F was determined from certified material test reports for each material in the flange region. The method used was that of NRC Branch Technical Position MTEB 5-2, Fracture Toughness Requirements, paragraph 1.1(3)(b), Reference 9. MTEB 5-2 provides a means of determining RT_{NDT} in cases such as this where testing requirements were established before the ASME Code requirements for RT_{NDT} (Section III, NB-2300) were issued.

The drop weight NDTT was established in accordance with ASTM E208. The Charpy specimens were oriented in the tangential orientation in the forgings and in the longitudinal orientation in the plates such that the impact energy and lateral expansion data were evaluated in accordance with MTEB 5-2, paragraph 1.1(3)(b). The RT_{NDT} for the welds was based on the generic value, -56 °F, for welds fabricated using Linde 1092 (et al.) flux following 10CFR50.61. The results of the data evaluation are summarized in the table below:

Flange Region Material Properties

Component	**Code No.**	**NDTT**	**T_{CV50} or T_{CV35}**
Vessel Flange	D-4803	0 °F	<30 °F
Closure Head.Flange	D-4805	-10 °F	<30 °F
Shell Plate	D-4801-1	-20 °F	44 °F
Shell Plate	D-4801-2	-30 °F	44 °F
Shell Plate	D-4801-3	10 °F	62 °F
Shell Girth Weld	7-410	N/A	N/A
Shell Axial Welds	1-410 A/C	N/A	N/A
Closure Head Torus	D-4808	-40 °F	50 °F
Cl. Head Girth Welds	2-415 A&B	N/A	N/A

The higher of the 50 ft-lb and 35 mils (lateral expansion) index temperatures was selected for each component; the higher value was T_{CV50} in all cases. The values shown were increased by 20 °F in accordance with MTEB 5-2, paragraph 1.1(3)(b), to obtain the value of T_{CV} adjusted to reflect the 'weak' orientation.

The ASME Section III, NB-2300 RT_{NDT} process was then followed with results given in the table below:

RT_{NDT} Determination for Flange Region Materials

Component	Code No.	NDTT	T_{CV} – 60 °F	RT_{NDT}
Vessel Flange	D-4803	0 °F	-30 °F	0 °F
Cl. Head. Flange	D-4805	-10 °F	-30 °F	-10 °F
Shell Plate	D-4801-1	-20 °F	-16 °F	-16 °F
Shell Plate	D-4801-2	-30 °F	-16 °F	-16°F
Shell Plate	D-4801-3	10 °F	+2 °F	10 °F
Girth Weld	7-410	N/A	N/A	-56 °F
Axial Welds	1-410 A/C	N/A	N/A	-56 °F
Cl. Head Torus	D-4808	-40 °F	-10 °F	-10 °F
Cl. Head Girth Welds	2-415 A&B	N/A	N/A	-56 °F

RT_{NDT} is the higher of NDTT and the quantity T_{CV} - 60 °F, and the highest RT_{NDT} for the flange region is +10 °F for the Fort Calhoun reactor vessel. Therefore, the minimum boltup temperature can be based on the RT_{NDT} for the flange region of +10 °F in accordance with the requirements of 10CFR50, Appendix G and thus needs no Code reconciliation to implement.

For conservatism, however, Reference 5 specified a bounding analysis value of 50°F. Using that value and applying the temperature correction, the minimum boltup temperature is 64°F.

4.6 Flange Limits

From Reference 1, the temperature of the closure flange regions must exceed the initial RT_{NDT} of the material by at least 120°F for normal operation and by 90°F for hydrostatic test and leak testing when the pressure exceeds twenty percent of pre-service hydrostatic test pressure.

Therefore, for normal operation the flange limit with instrumentation uncertainty is:

$Flange_{Normal\ Op}$ = Initial RT_{NDT} + 120°F + dT

= 10°F + 120°F + 14°F = 144°F

$Flange_{Hydro}$ = Initial RT_{NDT} + 90°F + dT
= 10°F + 90°F + 14°F = 114°F

These are the minimum temperatures for the flange regions in order for the pressure to exceed twenty percent of preservice hydrostatic test (625 psia uncorrected).

4.7 Minimum Pressure

This is the breakpoint between the minimum low pressure operating temperature and the lowest service temperature. The minimum pressure, defined by Reference 1 as twenty percent of the pre-service hydrostatic test pressure, is as follows:

20% of Preservice Hydrostatic Test =
(1.25 x Design Pressure) x 0.20 =
(1.25 x 2500 psia) x 0.20 = 625 psia

With pressure corrections due to flow and elevation (see Section 3.4), this becomes:

For T < 210°F, P = 625 psia – 61 psia = 564 psia

For T ≥ 210°F, P = 625 psia – 67 psia = 558 psia

4.8 Lowest Service Temperature

The lowest service temperature is defined in Reference 8 and is the minimum temperature for piping, pumps, and valves in the RCS that must exist to exceed twenty percent of pre-service hydrostatic test pressure.

Lowest Service Temperature =
Initial RT_{NDT} of piping, pumps and valves + 100°F

The initial RT_{NDT} for these components was conservatively established to be 50°F as follows. The ASME Code requirement for the determination of initial RT_{NDT} for pressure boundary materials, Section III, NB 2300, was established in the Summer 1972 Addenda. The Fort Calhoun vessel was fabricated before the Summer 1972 Addenda was issued. Therefore, a maximum estimated RT_{NDT} of 50°F was used for Fort Calhoun components (which pre-date required compliance with Section III, NB 2300). This is a reasonable value to use to determine the lowest service temperature for the carbon and

low alloy steel ferritic materials in the primary coolant system other than the reactor pressure vessel materials. The maximum estimated RT_{NDT} of 50°F is supported by an assessment of carbon and low alloy steel ferritic base materials in the reactor vessel, pressurizer, and primary piping that were designed and constructed to the 1968 Edition through Summer of 1969 Addenda of the ASME Code. Using NRC Branch Technical Position MTEB 5-2 methods to generate estimated RT_{NDT} values, the single highest value was 50°F and the remaining values were 30°F or less.

With instrument uncertainty included:

Lowest Service Temperature =
50°F + 100°F + 14°F = 164°F

4.9 LTOP Enable Temperatures

ASME Boiler and Pressure Vessel Code Section XI, Appendix G (Reference 3) has defined the LTOP systems to become effective at coolant temperatures less than 200°F or at coolant temperatures corresponding to reactor vessel temperatures less than RT_{NDT} + 50°F, whichever is greater.

The LTOP enable temperature for cooldown is based on the isothermal P-T limit. The LTOP enable temperature is, therefore, equal to the 1/4t adjusted reference temperature from Section 3.5 plus 50°F. Therefore, for cooldown, including instrumentation uncertainty;

LTOP Enable Temperature = 237.76 + 14 + 50 = 301.76 °F

For heatup, the LTOP enable temperatures along with the limiting location and temperature difference between the flaw tip location and the RCS fluid for the analyzed heatup rates are given below.

Heatup Rate (°F/hr)	Limiting Condition	LTOP Enable Temperature (°F)
10	1/4t	304.7
20	1/4t	307.7
30	1/4t	310.7
40	1/4t	313.6
50	1/4t	316.6
75	1/4t	324.0

Heatup Rate (°F/hr)	Limiting Condition	LTOP Enable Temperature (°F)
100	1/4t	331.4

The output from the PTCURVE code (see Section 4.1) was used to determine the above temperatures. These values include the temperature correction factor of 14°F.

Reference 5 directs that a value of 350°F be validated as an acceptable LTOP enable temperature. Comparison with the values in the table above shows that 350°F is acceptable.

4.10 Results

Results of these evaluations are presented in Tables 1 through 6 and Figures 1 through 8. Note that the curves presented in Figures 1 through 8 are for pictorial representation only and the data Tables are governing.

The beltline P-T limit data is tabulated in Tables 1 through 3 and presented in Figures 1 through 3. This information includes pressure and temperature correction factors. Pressure correction factors are due to flow and elevation only. The temperature correction factor is due to instrumentation uncertainty.

The beltline P-T limit data without pressure and temperature correction factors included is tabulated in Tables 4 through 6 and presented in Figures 4 through 6.

The composite corrected RCS P-T limits curves are presented in Figures 7 and 8. The hydrostatic test limits, core critical limits, minimum boltup temperature, minimum pressure, flange limits and lowest service temperature are presented in this report and are also included in the composite RCS P-T limit figures where appropriate. Pressure correction factors are due to flow and elevation only (no pressurizer pressure instrument uncertainty). The temperature correction factor is due to instrumentation uncertainty. As shown on both of these figures, the minimum pressure requirement plays a more significant role than in the past. For example, analysis of Figure 7 indicates that the minimum pressure requirements become more limiting during heatup at lower temperatures than the 75°F/hr curve.

LTOP enable temperatures are also presented in this report (Section 4.9). These temperatures were determined using the output from the PTCURVE computer code. It is also demonstrated that the calculated enable temperature values are less than the selected/preferred value of 350°F specified in Reference 5.

5.0 References

1. Title 10, Code of Federal Regulations, Part 50, Appendix G, Fracture Toughness Requirements, December 1995.
2. ASME Boiler and Pressure Vessel Code Section III, Appendix G, "Protection Against Nonductile Failure", 1986 Edition.
3. ASME Boiler and Pressure Vessel Code, 1995 Edition and addenda through the 1996 Addenda, Section XI, Appendix G, "Fracture Toughness Criteria for Protection Against Failure".
4. Cases of ASME Boiler and Pressure Vessel Code, Case N-640, "Alternative Reference Fracture Toughness for Development of P-T Limit Curves," Section XI, Division 1, dated February 26, 1996.
5. Letter No. NED-DEN-01-0147, Kevin C. Holthaus to Francis P. Ferraraccio, "Requested Input Data to be incorporated into the 40 EFPY Pressure and Temperature (P-T) Limit Curve Calculation", October 26, 2001.
6. ASME Boiler and Pressure Vessel Code, 1995 Edition and addenda through the 1996 Addenda, Section XI, Appendix A, "Analysis of Flaws."
7. "Semi-Elliptical Cracks in a Cylinder Subjected to Stress Gradients", J. Heliot, R.C. Labbens and Pellisser-Tanon ASTM Special Technical Publication 677, August 1979.
8. ASME Boiler and Pressure Vessel Code Section III, Article NB-2332, 1986 Edition.
9. NRC Branch Technical Position MTEB 5-2, "Fracture Toughness Requirements."

Table 1: OPPD Fort Calhoun Station Unit 1 P-Allowable vs. RCS Temperature, Heatup, Corrected

SRT Case//RCS HU PT CURVES-100//75/50/40/30/20/10 F/HR 40 EFPY
CORR TCOOLANT VS CORR P-ALL - Standard CE Method
PTCRV30B 11/01/2001 09:19:51

RCS	P-Allowable (ksi)							
Temp (°F)	Isothermal	10 F/hr	20 F/hr	30 F/hr	40 F/hr	50 F/hr	75 F/hr	100 F/hr
64	0.638	0.638	0.638	0.638	0.638	0.638	0.638	0.638
74	0.640	0.640	0.640	0.640	0.640	0.640	0.640	0.640
84	0.643	0.643	0.643	0.643	0.643	0.643	0.641	0.643
94	0.646	0.646	0.646	0.631	0.620	0.613	0.609	0.613
104	0.650	0.650	0.650	0.650	0.629	0.614	0.592	0.583
114	0.655	0.655	0.655	0.655	0.641	0.621	0.588	0.569
124	0.661	0.661	0.661	0.661	0.653	0.630	0.588	0.561
134	0.669	0.669	0.669	0.669	0.668	0.642	0.591	0.557
144	0.678	0.678	0.678	0.678	0.678	0.657	0.599	0.558
154	0.689	0.689	0.689	0.689	0.689	0.676	0.610	0.562
164	0.702	0.702	0.702	0.702	0.702	0.699	0.626	0.570
174	0.719	0.719	0.719	0.719	0.719	0.719	0.646	0.582
184	0.739	0.739	0.739	0.739	0.739	0.739	0.671	0.599
194	0.763	0.763	0.763	0.763	0.763	0.763	0.703	0.622
204	0.793	0.793	0.793	0.793	0.793	0.793	0.743	0.651
214	0.830	0.830	0.830	0.830	0.830	0.830	0.791	0.687
223.9	0.874	0.874	0.874	0.874	0.874	0.874	0.850	0.731
224	0.869	0.869	0.869	0.869	0.869	0.869	0.845	0.726
234	0.924	0.924	0.924	0.924	0.924	0.924	0.918	0.781
244	0.990	0.990	0.990	0.990	0.990	0.990	0.990	0.849
254	1.072	1.072	1.072	1.072	1.072	1.072	1.072	0.933
264	1.172	1.169	1.167	1.166	1.167	1.169	1.172	1.036
274	1.293	1.284	1.275	1.268	1.263	1.260	1.259	1.161
284	1.442	1.424	1.407	1.392	1.380	1.370	1.355	1.314
294	1.623	1.595	1.568	1.544	1.523	1.505	1.471	1.451
304	1.845	1.804	1.765	1.729	1.698	1.670	1.613	1.574
314	2.116	2.058	2.005	1.956	1.911	1.871	1.786	1.723
324	2.447	2.369	2.297	2.232	2.171	2.116	1.998	1.906
334	2.851	2.751	2.658	2.571	2.491	2.417	2.258	2.130
344	3.344	3.219	3.098	2.986	2.882	2.786	2.575	2.404
349.9	3.767	3.614	3.408	3.278	3.157	3.045	2.799	2.597
350	3.773	3.620	3.368	3.238	3.117	3.004	2.757	2.555
354	4.014	3.826	3.699	3.560	3.315	3.191	2.918	2.694
364	4.150	4.150	4.150	4.091	3.961	3.840	3.505	3.103
374	4.150	4.150	4.150	4.150	4.150	4.150	4.034	3.714
384	4.150	4.150	4.150	4.150	4.150	4.150	4.150	4.150
394	4.150	4.150	4.150	4.150	4.150	4.150	4.150	4.150
404	4.150	4.150	4.150	4.150	4.150	4.150	4.150	4.150
414	4.150	4.150	4.150	4.150	4.150	4.150	4.150	4.150
424	4.150	4.150	4.150	4.150	4.150	4.150	4.150	4.150
434	4.150	4.150	4.150	4.150	4.150	4.150	4.150	4.150
444	4.150	4.150	4.150	4.150	4.150	4.150	4.150	4.150
454	4.150	4.150	4.150	4.150	4.150	4.150	4.150	4.150
464	4.150	4.150	4.150	4.150	4.150	4.150	4.150	4.150
474	4.150	4.150	4.150	4.150	4.150	4.150	4.150	4.150
484	4.150	4.150	4.150	4.150	4.150	4.150	4.150	4.150
494	4.150	4.150	4.150	4.150	4.150	4.150	4.150	4.150
504	4.150	4.150	4.150	4.150	4.150	4.150	4.150	4.150
514	4.150	4.150	4.150	4.150	4.150	4.150	4.150	4.150
524	4.150	4.150	4.150	4.150	4.150	4.150	4.150	4.150
534	4.150	4.150	4.150	4.150	4.150	4.150	4.150	4.150
544	4.150	4.150	4.150	4.150	4.150	4.150	4.150	4.150
554	4.150	4.150	4.150	4.150	4.150	4.150	4.150	4.150
564	4.150	4.150	4.150	4.150	4.150	4.150	4.150	4.150

Table 2: OPPD Fort Calhoun Station Unit 1 P-Allowable vs. RCS Temperature, Cooldown, Corrected

SRT Case//RCS COOLDOWN PT CURVES-100//75/50/40/30/20/10 F/HR 40 EFPY
CORR TCOOLANT VS CORR P-ALL - Standard CE Method
PTCRV30B 11/01/2001 09:09:41

RCS	P-Allowable (ksi)							
Temp (°F)	Isothermal	100 F/hr	75 F/hr	50 F/hr	40 F/hr	30 F/hr	20 F/hr	10 F/hr
64	0.638	0.360	0.429	0.498	0.526	0.554	0.582	0.610
74	0.640	0.364	0.433	0.502	0.529	0.557	0.585	0.612
84	0.643	0.369	0.437	0.505	0.533	0.560	0.588	0.615
94	0.646	0.376	0.442	0.510	0.537	0.564	0.591	0.619
104	0.650	0.383	0.449	0.515	0.542	0.569	0.596	0.623
114	0.655	0.392	0.457	0.522	0.548	0.575	0.602	0.628
124	0.661	0.404	0.466	0.530	0.556	0.582	0.608	0.635
134	0.669	0.417	0.478	0.540	0.566	0.591	0.617	0.643
144	0.678	0.434	0.492	0.553	0.577	0.602	0.627	0.652
154	0.689	0.455	0.510	0.568	0.591	0.615	0.640	0.664
164	0.702	0.480	0.531	0.586	0.609	0.632	0.655	0.678
174	0.719	0.510	0.558	0.609	0.630	0.651	0.673	0.696
184	0.739	0.548	0.590	0.636	0.655	0.676	0.696	0.717
194	0.763	0.593	0.629	0.669	0.687	0.705	0.724	0.743
204	0.793	0.649	0.676	0.710	0.725	0.741	0.758	0.775
214	0.830	0.717	0.735	0.760	0.772	0.785	0.799	0.814
223.9	0.874	0.799	0.805	0.821	0.829	0.839	0.850	0.862
224	0.869	0.794	0.800	0.815	0.824	0.833	0.844	0.856
234	0.924	0.895	0.887	0.890	0.894	0.899	0.906	0.914
244	0.990	0.990	0.990	0.981	0.979	0.979	0.982	0.985
254	1.072	1.072	1.072	1.072	1.072	1.072	1.072	1.072
264	1.172	1.172	1.172	1.172	1.172	1.172	1.172	1.172
274	1.293	1.293	1.293	1.293	1.293	1.293	1.293	1.293
284	1.442	1.442	1.442	1.442	1.442	1.442	1.442	1.442
294	1.623	1.623	1.623	1.623	1.623	1.623	1.623	1.623
304	1.845	1.845	1.845	1.845	1.845	1.845	1.845	1.845
314	2.116	2.116	2.116	2.116	2.116	2.116	2.116	2.116
324	2.447	2.447	2.447	2.447	2.447	2.447	2.447	2.447
334	2.851	2.851	2.851	2.851	2.851	2.851	2.851	2.851
344	3.344	3.344	3.344	3.344	3.344	3.344	3.344	3.344
349.9	3.767	3.767	3.767	3.767	3.767	3.767	3.767	3.767
350	3.773	3.773	3.773	3.773	3.773	3.773	3.773	3.773
354	4.014	3.867	3.936	4.007	4.014	4.014	4.014	4.014
364	4.150	3.868	3.936	4.007	4.036	4.064	4.093	4.122
374	4.150	3.869	3.937	4.007	4.036	4.064	4.093	4.122
384	4.150	3.870	3.937	4.007	4.036	4.064	4.093	4.122
394	4.150	3.872	3.937	4.007	4.036	4.064	4.093	4.122
404	4.150	3.874	3.938	4.007	4.036	4.064	4.093	4.122
414	4.150	3.876	3.939	4.007	4.036	4.064	4.093	4.122
424	4.150	3.879	3.940	4.007	4.036	4.064	4.093	4.122
434	4.150	3.883	3.941	4.008	4.036	4.064	4.093	4.122
444	4.150	3.888	3.943	4.008	4.036	4.064	4.093	4.122
454	4.150	3.893	3.946	4.009	4.036	4.064	4.093	4.122
464	4.150	3.900	3.949	4.009	4.036	4.064	4.093	4.122
474	4.150	3.909	3.953	4.010	4.037	4.064	4.093	4.122
484	4.150	3.920	3.959	4.012	4.037	4.065	4.093	4.122
494	4.150	3.933	3.967	4.015	4.039	4.065	4.093	4.122
504	4.150	3.949	3.977	4.019	4.041	4.066	4.093	4.122
514	4.150	3.970	3.991	4.026	4.045	4.067	4.093	4.122
524	4.150	3.995	4.010	4.035	4.051	4.070	4.094	4.122
534	4.150	4.026	4.034	4.050	4.061	4.076	4.096	4.122
544	4.150	4.065	4.066	4.073	4.079	4.087	4.101	4.122
554	4.150	4.110	4.109	4.108	4.108	4.111	4.115	4.126
564	4.150	4.150	4.150	4.150	4.150	4.150	4.150	4.150

Table 3: OPPD Fort Calhoun Station Unit 1 P-Allowable vs. RCS Temperature, Hydrostatic Operation, Corrected

SRT Case//RCS HYDROSTATIC PT CURVES - 40 EFPY
CORR TCOOLANT VS CORR P-ALL - Standard CE Method
PTCRV30B 11/01/2001 09:14:07

RCS	P-Allowable (ksi)	
Temp (°F)	Isothermal	100 F/hr
64	0.871	0.501
74	0.874	0.506
84	0.878	0.513
94	0.882	0.521
104	0.887	0.531
114	0.894	0.544
124	0.902	0.559
134	0.912	0.577
144	0.924	0.599
154	0.939	0.627
164	0.957	0.660
174	0.979	0.701
184	1.006	0.750
194	1.038	0.811
204	1.078	0.885
214	1.127	0.976
223.9	1.186	1.086
224	1.181	1.081
234	1.254	1.216
244	1.343	1.343
254	1.452	1.452
264	1.584	1.584
274	1.747	1.747
284	1.945	1.945
294	2.187	2.187
304	2.482	2.482
314	2.843	2.843
324	3.284	3.284
334	3.890	3.890
344	4.548	4.548
349.9	5.022	5.022
350	5.030	5.030
354	5.352	5.156
364	5.534	5.158
374	5.534	5.159
384	5.534	5.161
394	5.534	5.163
404	5.534	5.165
414	5.534	5.169
424	5.534	5.172
434	5.534	5.177
444	5.534	5.183
454	5.534	5.191
464	5.534	5.200
474	5.534	5.212
484	5.534	5.226
494	5.534	5.244
504	5.534	5.266
514	5.534	5.293
524	5.534	5.327
534	5.534	5.369
544	5.534	5.420
554	5.534	5.481
564	5.534	5.534

Table 4: OPPD Fort Calhoun Station Unit 1 P-Allowable vs. RCS Temperature, Heatup, Uncorrected

SRT Case//RCS HU PT CURVES-100//75/50/40/30/20/10 F/HR 40 EFPY
UNCORR TCOOLANT VS UNCORR P-ALL - Standard CE Method
PTCRV30B 11/01/2001 09:19:51

RCS	P-Allowable (ksi)							
Temp (°F)	Isothermal	10 F/hr	20 F/hr	30 F/hr	40 F/hr	50 F/hr	75 F/hr	100 F/hr
50	0.699	0.699	0.699	0.699	0.699	0.699	0.699	0.699
60	0.701	0.701	0.701	0.701	0.701	0.701	0.701	0.701
70	0.704	0.704	0.704	0.704	0.704	0.704	0.702	0.704
80	0.707	0.707	0.707	0.692	0.681	0.674	0.670	0.674
90	0.711	0.711	0.711	0.711	0.690	0.675	0.653	0.644
100	0.716	0.716	0.716	0.716	0.702	0.682	0.649	0.630
110	0.722	0.722	0.722	0.722	0.714	0.691	0.649	0.622
120	0.730	0.730	0.730	0.730	0.729	0.703	0.652	0.618
130	0.739	0.739	0.739	0.739	0.739	0.718	0.660	0.619
140	0.750	0.750	0.750	0.750	0.750	0.737	0.671	0.623
150	0.763	0.763	0.763	0.763	0.763	0.760	0.687	0.631
160	0.780	0.780	0.780	0.780	0.780	0.780	0.707	0.643
170	0.800	0.800	0.800	0.800	0.800	0.800	0.732	0.660
180	0.824	0.824	0.824	0.824	0.824	0.824	0.764	0.683
190	0.854	0.854	0.854	0.854	0.854	0.854	0.804	0.712
200	0.891	0.891	0.891	0.891	0.891	0.891	0.852	0.748
209.9	0.935	0.935	0.935	0.935	0.935	0.935	0.911	0.792
210	0.936	0.936	0.936	0.936	0.936	0.936	0.912	0.793
220	0.991	0.991	0.991	0.991	0.991	0.991	0.985	0.848
230	1.057	1.057	1.057	1.057	1.057	1.057	1.057	0.916
240	1.139	1.139	1.139	1.139	1.139	1.139	1.139	1.000
250	1.239	1.236	1.234	1.233	1.234	1.236	1.239	1.103
260	1.360	1.351	1.342	1.335	1.330	1.327	1.326	1.228
270	1.509	1.491	1.474	1.459	1.447	1.437	1.422	1.381
280	1.690	1.662	1.635	1.611	1.590	1.572	1.538	1.518
290	1.912	1.871	1.832	1.796	1.765	1.737	1.680	1.641
300	2.183	2.125	2.072	2.023	1.978	1.938	1.853	1.790
310	2.514	2.436	2.364	2.299	2.238	2.183	2.065	1.973
320	2.918	2.818	2.725	2.638	2.558	2.484	2.325	2.197
330	3.411	3.286	3.165	3.053	2.949	2.853	2.642	2.471
335.9	3.767	3.614	3.475	3.345	3.224	3.112	2.866	2.664
336	3.773	3.620	3.480	3.350	3.229	3.116	2.869	2.667
340	4.014	3.826	3.699	3.560	3.427	3.303	3.030	2.806
350	4.150	4.150	4.150	4.091	3.961	3.840	3.505	3.215
360	4.150	4.150	4.150	4.150	4.150	4.150	4.034	3.714
370	4.150	4.150	4.150	4.150	4.150	4.150	4.150	4.150
380	4.150	4.150	4.150	4.150	4.150	4.150	4.150	4.150
390	4.150	4.150	4.150	4.150	4.150	4.150	4.150	4.150
400	4.150	4.150	4.150	4.150	4.150	4.150	4.150	4.150
410	4.150	4.150	4.150	4.150	4.150	4.150	4.150	4.150
420	4.150	4.150	4.150	4.150	4.150	4.150	4.150	4.150
430	4.150	4.150	4.150	4.150	4.150	4.150	4.150	4.150
440	4.150	4.150	4.150	4.150	4.150	4.150	4.150	4.150
450	4.150	4.150	4.150	4.150	4.150	4.150	4.150	4.150
460	4.150	4.150	4.150	4.150	4.150	4.150	4.150	4.150
470	4.150	4.150	4.150	4.150	4.150	4.150	4.150	4.150
480	4.150	4.150	4.150	4.150	4.150	4.150	4.150	4.150
490	4.150	4.150	4.150	4.150	4.150	4.150	4.150	4.150
500	4.150	4.150	4.150	4.150	4.150	4.150	4.150	4.150
510	4.150	4.150	4.150	4.150	4.150	4.150	4.150	4.150
520	4.150	4.150	4.150	4.150	4.150	4.150	4.150	4.150
530	4.150	4.150	4.150	4.150	4.150	4.150	4.150	4.150
540	4.150	4.150	4.150	4.150	4.150	4.150	4.150	4.150
550	4.150	4.150	4.150	4.150	4.150	4.150	4.150	4.150

Table 5: OPPD Fort Calhoun Station Unit 1 P-Allowable vs. RCS Temperature, Cooldown, Uncorrected

SRT Case//RCS COOLDOWN PT CURVES-100//75/50/40/30/20/10 F/HR 40 EFPY
UNCORR TCOOLANT VS UNCORR P-ALL - Standard CE Method
PTCRV30B 11/01/2001 09:09:41

RCS	P-Allowable (ksi)							
Temp (°F)	Isothermal	100 F/hr	75 F/hr	50 F/hr	40 F/hr	30 F/hr	20 F/hr	10 F/hr
50	0.699	0.421	0.490	0.559	0.587	0.615	0.643	0.671
60	0.701	0.425	0.494	0.563	0.590	0.618	0.646	0.673
70	0.704	0.430	0.498	0.566	0.594	0.621	0.649	0.676
80	0.707	0.437	0.503	0.571	0.598	0.625	0.652	0.680
90	0.711	0.444	0.510	0.576	0.603	0.630	0.657	0.684
100	0.716	0.453	0.518	0.583	0.609	0.636	0.663	0.689
110	0.722	0.465	0.527	0.591	0.617	0.643	0.669	0.696
120	0.730	0.478	0.539	0.601	0.627	0.652	0.678	0.704
130	0.739	0.495	0.553	0.614	0.638	0.663	0.688	0.713
140	0.750	0.516	0.571	0.629	0.652	0.676	0.701	0.725
150	0.763	0.541	0.592	0.647	0.670	0.693	0.716	0.739
160	0.780	0.571	0.619	0.670	0.691	0.712	0.734	0.757
170	0.800	0.609	0.651	0.697	0.716	0.737	0.757	0.778
180	0.824	0.654	0.690	0.730	0.748	0.766	0.785	0.804
190	0.854	0.710	0.737	0.771	0.786	0.802	0.819	0.836
200	0.891	0.778	0.796	0.821	0.833	0.846	0.860	0.875
209.9	0.935	0.860	0.866	0.882	0.890	0.900	0.911	0.923
210	0.936	0.861	0.867	0.882	0.891	0.900	0.911	0.923
220	0.991	0.962	0.954	0.957	0.961	0.966	0.973	0.981
230	1.057	1.057	1.057	1.048	1.046	1.046	1.049	1.052
240	1.139	1.139	1.139	1.139	1.139	1.139	1.139	1.139
250	1.239	1.239	1.239	1.239	1.239	1.239	1.239	1.239
260	1.360	1.360	1.360	1.360	1.360	1.360	1.360	1.360
270	1.509	1.509	1.509	1.509	1.509	1.509	1.509	1.509
280	1.690	1.690	1.690	1.690	1.690	1.690	1.690	1.690
290	1.912	1.912	1.912	1.912	1.912	1.912	1.912	1.912
300	2.183	2.183	2.183	2.183	2.183	2.183	2.183	2.183
310	2.514	2.514	2.514	2.514	2.514	2.514	2.514	2.514
320	2.918	2.918	2.918	2.918	2.918	2.918	2.918	2.918
330	3.411	3.411	3.411	3.411	3.411	3.411	3.411	3.411
335.9	3.767	3.767	3.767	3.767	3.767	3.767	3.767	3.767
336	3.773	3.773	3.773	3.773	3.773	3.773	3.773	3.773
340	4.014	3.867	3.936	4.007	4.014	4.014	4.014	4.014
350	4.150	3.868	3.936	4.007	4.036	4.064	4.093	4.122
360	4.150	3.869	3.937	4.007	4.036	4.064	4.093	4.122
370	4.150	3.870	3.937	4.007	4.036	4.064	4.093	4.122
380	4.150	3.872	3.937	4.007	4.036	4.064	4.093	4.122
390	4.150	3.874	3.938	4.007	4.036	4.064	4.093	4.122
400	4.150	3.876	3.939	4.007	4.036	4.064	4.093	4.122
410	4.150	3.879	3.940	4.007	4.036	4.064	4.093	4.122
420	4.150	3.883	3.941	4.008	4.036	4.064	4.093	4.122
430	4.150	3.888	3.943	4.008	4.036	4.064	4.093	4.122
440	4.150	3.893	3.946	4.009	4.036	4.064	4.093	4.122
450	4.150	3.900	3.949	4.009	4.036	4.064	4.093	4.122
460	4.150	3.909	3.953	4.010	4.037	4.064	4.093	4.122
470	4.150	3.920	3.959	4.012	4.037	4.065	4.093	4.122
480	4.150	3.933	3.967	4.015	4.039	4.065	4.093	4.122
490	4.150	3.949	3.977	4.019	4.041	4.066	4.093	4.122
500	4.150	3.970	3.991	4.026	4.045	4.067	4.093	4.122
510	4.150	3.995	4.010	4.035	4.051	4.070	4.094	4.122
520	4.150	4.026	4.034	4.050	4.061	4.076	4.096	4.122
530	4.150	4.065	4.066	4.073	4.079	4.087	4.101	4.122
540	4.150	4.110	4.109	4.108	4.108	4.111	4.115	4.126
550	4.150	4.150	4.150	4.150	4.150	4.150	4.150	4.150

Table 6: OPPD Fort Calhoun Station Unit 1 P-Allowable vs. RCS Temperature, Hydrostatic Operation, Uncorrected

SRT Case//RCS HYDROSTATIC PT CURVES - 40 EFPY
UNCORR TCOOLANT VS UNCORR P-ALL - Standard CE Method
PTCRV30B 11/01/2001 09:14:07

RCS	P-Allowable (ksi)	
Temp (°F)	Isothermal	100 F/hr
50	0.932	0.562
60	0.935	0.567
70	0.939	0.574
80	0.943	0.582
90	0.948	0.592
100	0.955	0.605
110	0.963	0.620
120	0.973	0.638
130	0.985	0.660
140	1.000	0.688
150	1.018	0.721
160	1.040	0.762
170	1.067	0.811
180	1.099	0.872
190	1.139	0.946
200	1.188	1.037
209.9	1.247	1.147
210	1.248	1.148
220	1.321	1.283
230	1.410	1.410
240	1.519	1.519
250	1.651	1.651
260	1.814	1.814
270	2.012	2.012
280	2.254	2.254
290	2.549	2.549
300	2.910	2.910
310	3.351	3.351
320	3.890	3.890
330	4.548	4.548
335.9	5.022	5.022
336	5.030	5.030
340	5.352	5.156
350	5.534	5.158
360	5.534	5.159
370	5.534	5.161
380	5.534	5.163
390	5.534	5.165
400	5.534	5.169
410	5.534	5.172
420	5.534	5.177
430	5.534	5.183
440	5.534	5.191
450	5.534	5.200
460	5.534	5.212
470	5.534	5.226
480	5.534	5.244
490	5.534	5.266
500	5.534	5.293
510	5.534	5.327
520	5.534	5.369
530	5.534	5.420
540	5.534	5.481
550	5.534	5.534

Figure 1: OPPD Fort Calhoun Station Unit 1, Beltline P-T Limits, Heatup

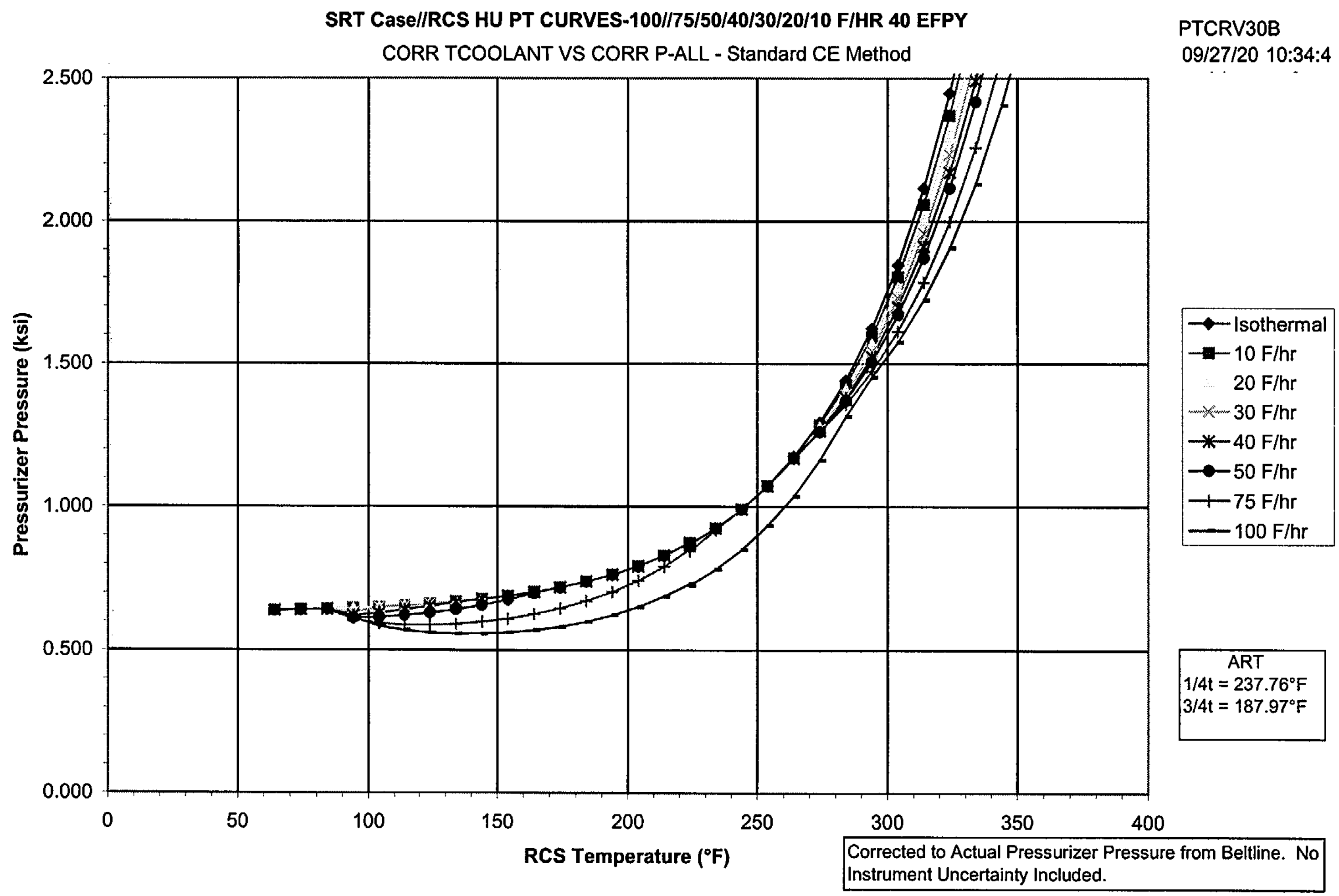


Figure 2: OPPD Fort Calhoun Station Unit 1, Beltline P-T Limits, Cooldown

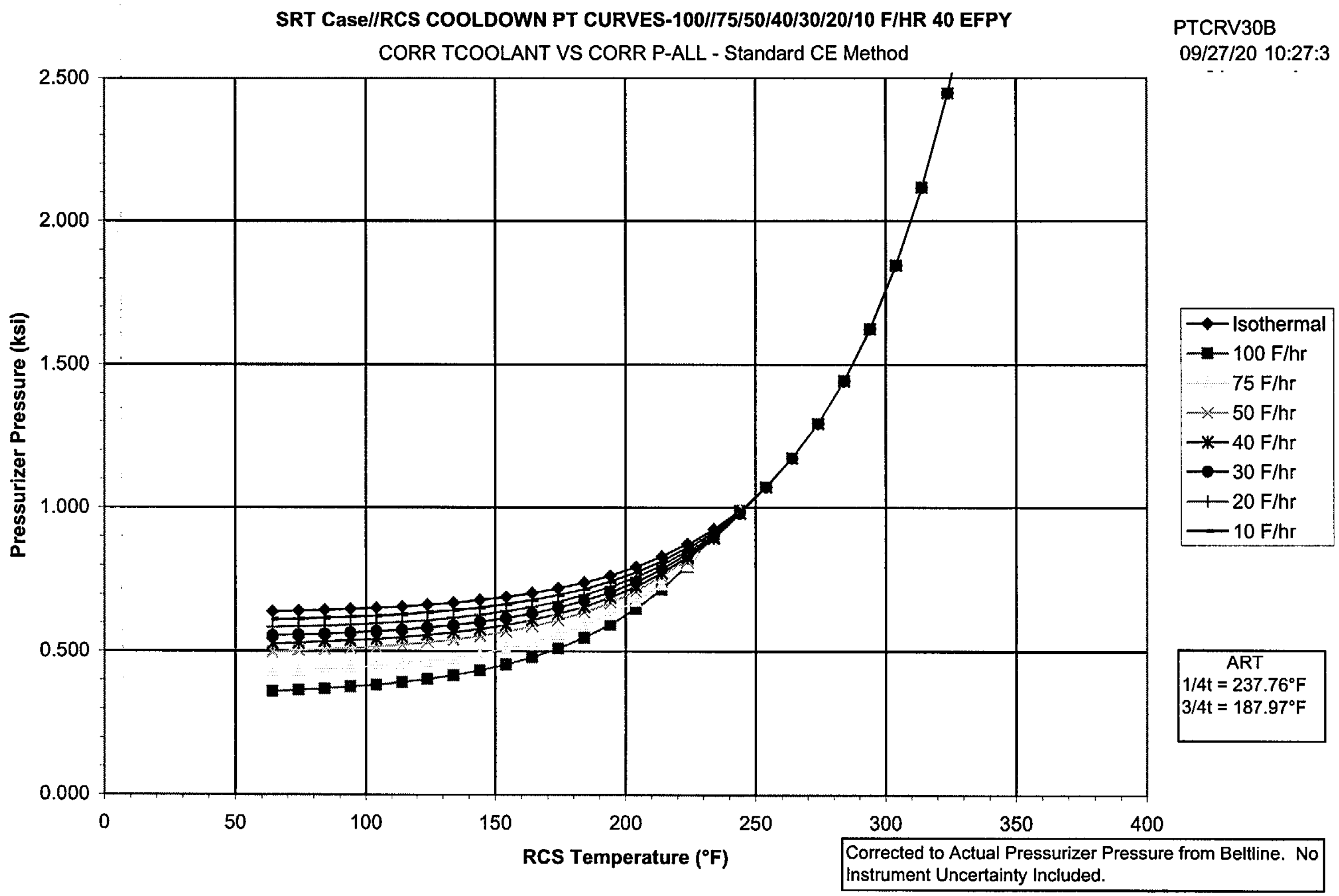


Figure 3: OPPD Fort Calhoun Station Unit 1, Beltline P-T Limits, Hydrostatic

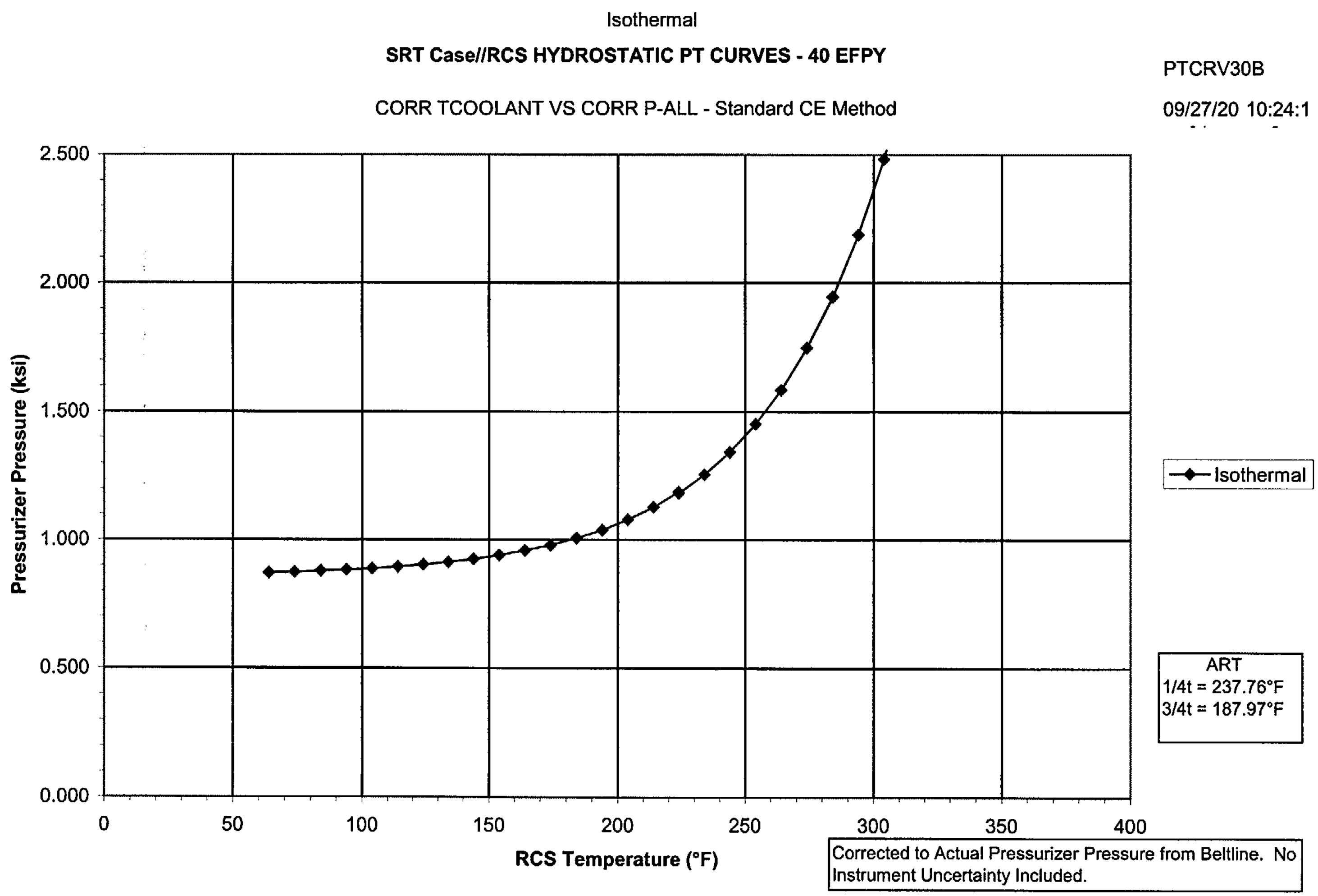


Figure 4: OPPD Fort Calhoun Station Unit 1, P-T Limits Uncorrected, Heatup

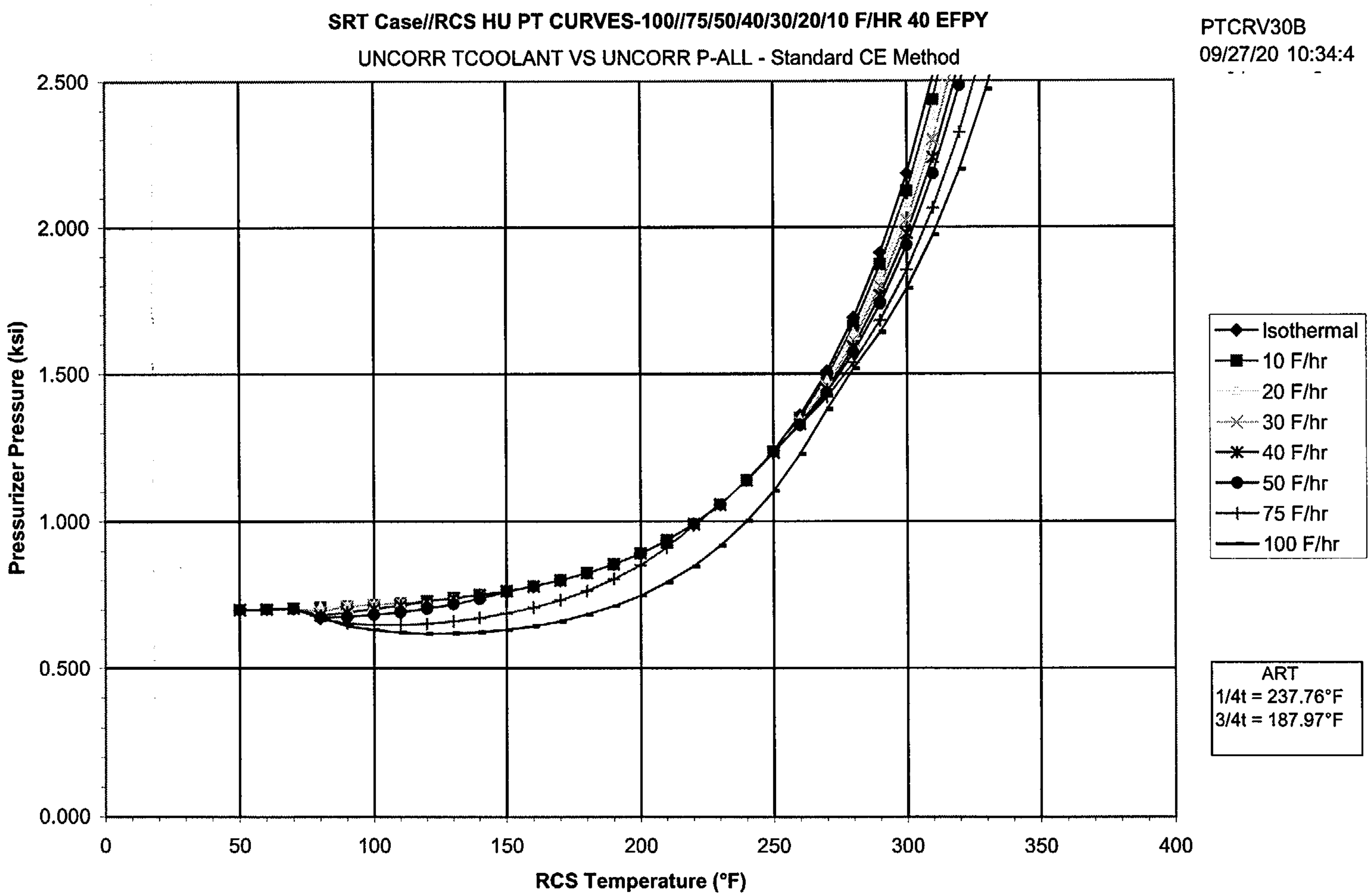


Figure 5: OPPD Fort Calhoun Station Unit 1, P-T Limits Uncorrected, Cooldown

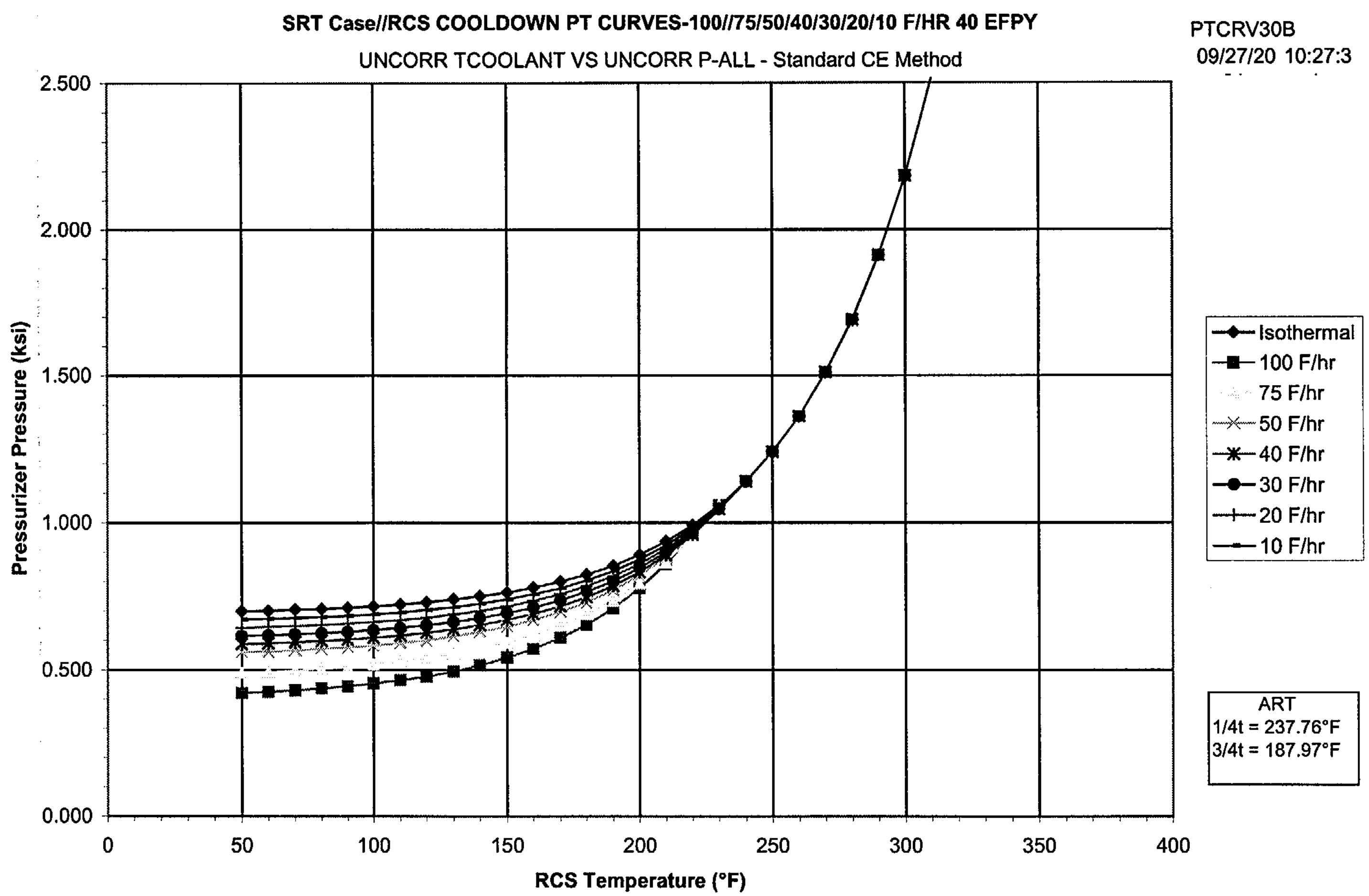


Figure 6: OPPD Fort Calhoun Station Unit 1, P-T Limits Uncorrected, Hydrostatic

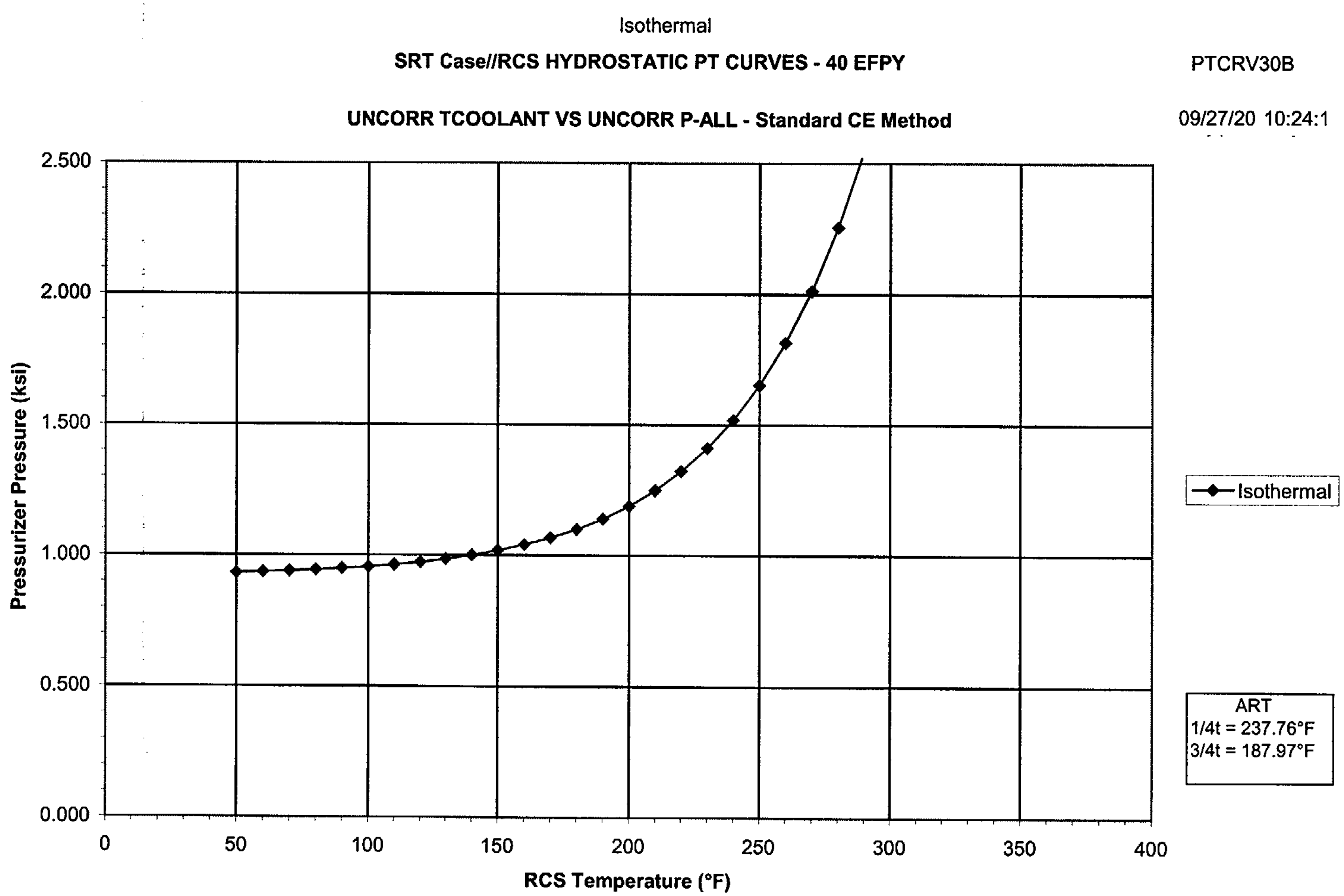


Figure 7: OPPD Fort Calhoun Station Unit 1, Composite P-T Limits, 40 EFPY (K_{IC}) Heatup

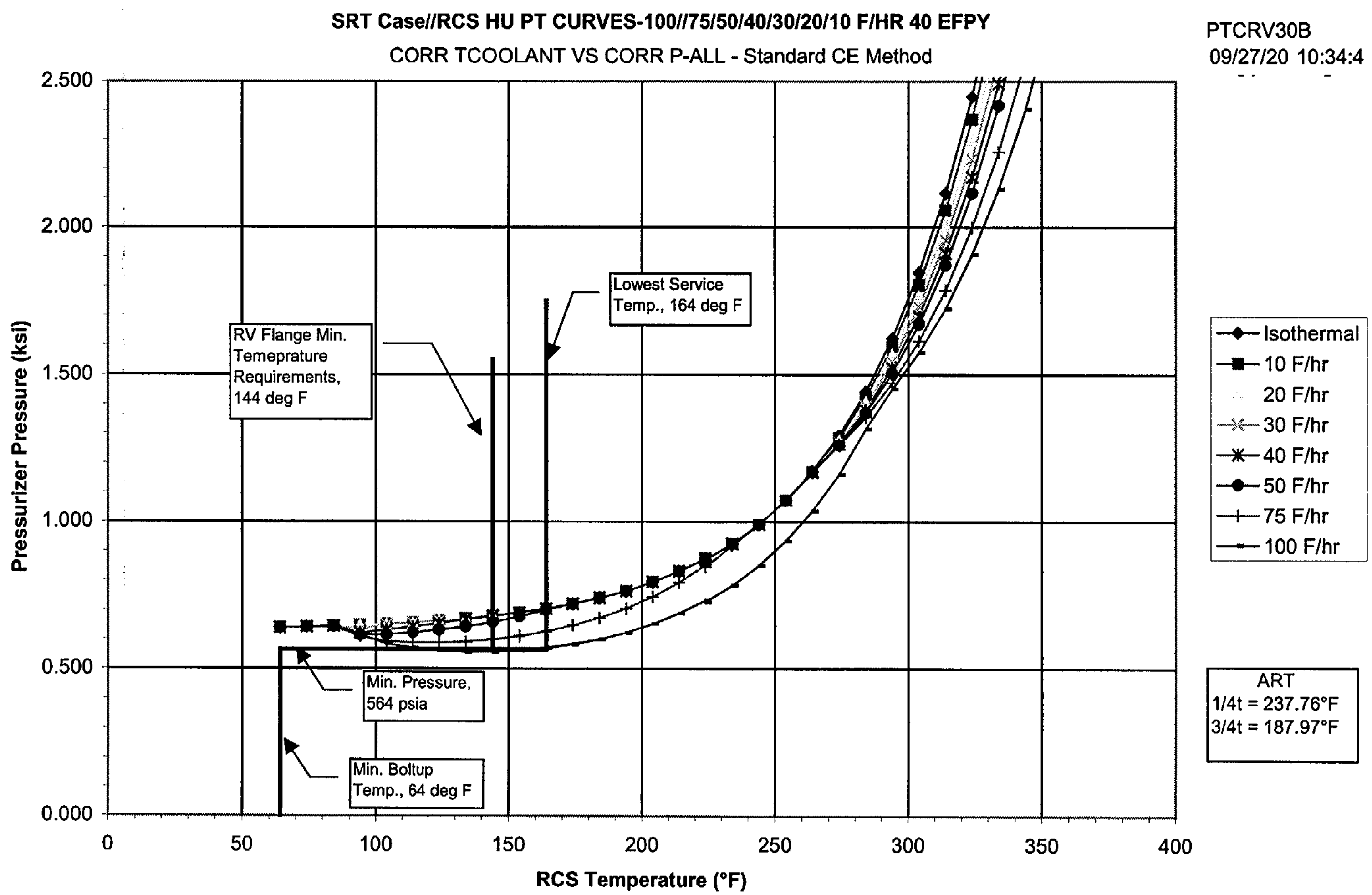
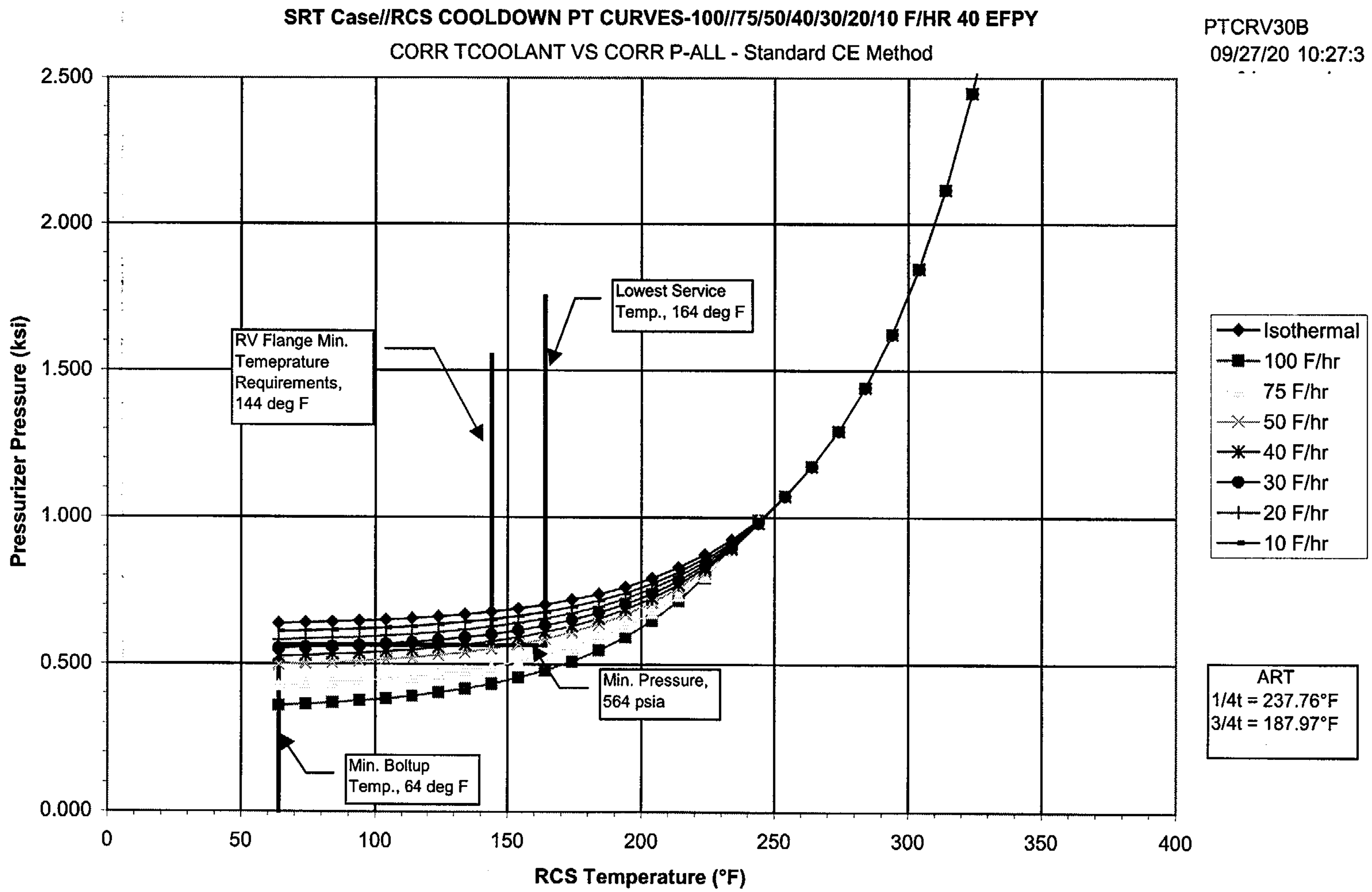


Figure 8: OPPD Fort Calhoun Station Unit 1, Composite P-T Limits, 40 EFPY (K_{IC}) Cooldown



References for Cover Letter

1. Docket No. 50-285
2. WCAP-15443, Revision 0, "Fast Neutron Fluence Evaluations for the Fort Calhoun Unit 1 Reactor Pressure Vessel," dated July 2000
3. Letter from NRC (A. B. Wang) to OPPD (S. K. Gambhir), dated June 6, 2001,"Fort Calhoun Station Unit No. 1 - Issuance of Amendment - Deletion of Section 3.D, "License Term" (TAC No. MA9690)" (NRC-01-058)
4. ASME Boiler and Pressure Vessel Code, 1986 Edition
5. ASME Boiler and Pressure Vessel Code, 1995 Edition and Addenda through the 1996 Addenda
6. Letter from OPPD (S. K. Gambhir) to NRC (Document Control Desk), dated December 14, 2001, Fort Calhoun Station Unit No. 1 Exemption Request,"10 CFR Part 50, Appendix G, Requirements" (LIC-01-0122)
7. Letter from Donna Skay, Project Manager, U.S. Nuclear Regulatory Commission, to C. Cruse, Vice President-Nuclear Energy, Calvert Cliffs Nuclear Power Plant, Inc., "Calvert Cliffs Nuclear Power Plant, Unit Nos. 1 and 2 - Amendment Re: Pressure-Temperature Curves (TAC Nos. MA9999 and MB0000)," dated March 15, 2001
8. Safety Evaluation of Topical Report CE NPSD-683, Revision 6, "Development of a RCS Pressure and Temperature Limits Report (PTLR) for the Removal of P-T Limits and LTOP Requirements from the Technical Specifications" (TAC No. MA9561)
9. Letter from G. F. Wunder, Project Manager, U.S. Nuclear Regulatory Commission, to J. Knubel, Senior Vice President, New York Power Authority, "Exemption from the Requirements of 10 CFR 50.60, 'Acceptance Criteria for Fracture Prevention Measures for Lightwater Nuclear Power Reactors for Normal Operation,' to Allow for Use of Alternative Methodology for Construction of Pressure Temperature Limit Curves - Indian Point Nuclear Generating Unit No. 3 (TAC No. M99928)," April 10, 1998