Fort Calhoun Station
Unit No. 1

TDB-IX

TECHNICAL DATA BOOK

Title: RCS PRESSURE - TEMPERATURE LIMITS REPORT (PTLR)

FC-68 Number:

Reason for Change:

Contact Person:

ISSUED: May 17, 1999 (2:39PM) R0

Table of Contents

1. INTRODUCTION

This PTLR for Fort Calhoun Station (FCS) Unit No. 1 contains Pressure-Temperature (P-T) limits corresponding to 20 Effective Full Power Years (EFPY) of operation. In addition, this report contains Low Temperature Overpressure Protection (LTOP) specific requirements which have been developed to protect the P-T limits from being exceeded during the limiting LTOP event.

The Technical Specifications affected by this report are listed below and are separated into the appropriate category; P-T limits or LTOP requirements.

2. GL 96-03 (Reference 3.1) PROVISION REQUIREMENTS

2.1 Neutron Fluence Values

The reactor vessel beltline neutron fluence has been calculated for the critical locations using the general methodology outline described in Reference 3.2, along with the details of the Fort Calhoun Station specific calculational methodology contained in Reference 3.7, which also includes evaluations of the surveillance capsule dosimetry for the W-225, W-265, and W-275 capsules.

Since the time of the performance of the analyses supporting the 20 EFPY limits in References 3.3, 3.15 and 3.16 several significant operating and analysis changes have occurred at FCS. The Reference 3.3 (based upon References 3.15 and 3.16) 20 EFPY limit analysis was performed in 1990 at a time when low radial leakage fuel management was being used as well as the ENDF/B-IV nuclear cross-section library. The fast neutron fluence (E> 1 MeV) at 20 EFPY was projected to be 1.501×10^{19} n/cm^2, to the limiting reactor vessel material (i.e., the 3-410 welds at 60° and 300° and up to approximately 40% of core height). It should be noted that although there is a 3-410 weld at 180° the fluence at this location is significantly less than at the 60° and 300° locations and it is therefore not limiting. In Cycle 14 extreme low radial leakage fuel management was implemented to further reduce the reactor vessel fast neutron flux and insure operation to August 9, 2013 without exceeding the 10CFR50.61 PTS screening criteria. Use of the ENDF/B-VI cross-section library in a 1994 updated fluence analysis (Reference 3.7) in conjunction with methods described in References 3.4 and 3.5 (and its predecessor Reference 3.6) resulted in a revised projection at 20 EFPY of 1.150×10^{19} n/cm^2. Since the original analysis remains conservative with respect to the actual projected fast neutron fluence, the Reference 3.3 results are utilized herein. It should be noted that further conservatism exists in the reactor vessel weld chemistry factor of 234.5°F used in References 3.3, 3.15 and 3.16 versus the most recently derived value of 231.06°F (per Reference 3.8).

2.1 In References 3.3, 3.15 and 3.16 the peak value(s) of fast neutron fluence (E > 1 MeV) at the vessel clad interface was used as input to the Adjusted Reference Temperature (ART) calculations for FCS. The fluence corresponding to the limiting 3-410 welds located at 60° and 300° for 20 effective full power years (EFPY) is 1.501×10^{19} neutrons per square centimeter (n/cm^2) with an associated uncertainty of less than ± 20%. In the updated Reference 3.7 analysis the uncertainty was shown to be less than ± 13%.

2.2 Reactor Vessel Surveillance Program

The reactor vessel surveillance program was developed in accordance with ASTM-E-185-66. The surveillance capsule withdrawal details and schedule are described in Reference 3.9 (and Reference 3.2). The reports describing the post-irradiation evaluation of the surveillance capsules are contained in References 3.10, 3.11 and 3.12. Each removed capsule has been evaluated in accordance with the testing requirements of the version of ASTM-E-185 in effect at the time of capsule removal. Reference 3.13 describes the pre-irradiation materials baseline surveillance program.

2.3 LTOP System Limits

The LTOP requirements have been developed by making a comparison between the peak transient pressures and the appropriate Appendix G pressure-temperature limit curves. The acceptability criterion regarding each particular transient is that the peak transient pressure does not exceed the applicable Appendix G pressure limit. These requirements for LTOP have been established based on NRC-accepted methodologies and are described in Reference 3.2, Section 3.0.

The affected Technical Specification Limiting Conditions for Operation (LCO's) which ensure adequate LTOP are:

T.S. 2.1.1 Reactor Coolant System - Operable Components

T.S. 2.1.2 Reactor Coolant System - Heatup and Cooldown Rate

T.S. 2.1.6 Reactor Coolant System - Pressurizer and Main Steam Safety Valves

T.S. 2.2.1 Chemical and Volume Control System - Boric Acid Flow Paths - Shutdown

T.S. 2.2.2 Chemical and Volume Control System - Boric Acid Flow Paths - Operating

2.3

T.S. 2.2.3	Chemical and Volume Control System - Charging Pumps - Shutdown
T.S. 2.2.4	Chemical and Volume Control System - Charging Pumps - Operating
T.S. 2.2.5	Chemical and Volume Control System - Boric Acid Transfer Pumps - Shutdown
T.S. 2.2.6	Chemical and Volume Control System - Boric Acid Transfer Pumps - Operating
T.S. 2.2.7	Chemical and Volume Control System - Borated Water Source - Shutdown
T.S. 2.2.8	Chemical and Volume Control System - Borated Water Sources - Operating
T.S. 2.3(3)	Emergency Core Cooling System - Protection Against Low Temperature Overpressurization

The LTOP specific requirements for each T.S. LCO, when applicable, are presented in the following subsections.

2.3.1 Reactor Coolant System (T.S. 2.1)

A. Operable Components (T.S. 2.1.1)

The following requirements apply:

1) Reactor coolant system leak and hydrostatic tests shall be conducted within the limitations of Figure 4.2.

2) If no reactor coolant pumps are operating a reactor pump shall not be started while T_c is below 385° F (LTOP enable temperature) unless at least one of the following is met:

(i) A pressurizer steam space of 53% by volume (50.6% or less actual level) exists, or

(ii) The steam generator secondary side temperature is less than 30°F above that of the reactor coolant system cold leg.

Startup of the first RCP with "cold steam generators", under these conditions may result in a cooldown which exceeds the 10°F/hour cooldown limit of Figure 4.1, however per the analysis of Reference 3.14 this does not result in exceeding the ASME Section III (and XI) or 10CFR50 Appendix G criteria. Thus, startup of the first RCP, which is considered to be a normal plant evolution, with a resultant step cooldown of up to 50°F is acceptable, provided the pressurizer pressure remains less than 275 psia. Reference 3.14 (page B5) shows the allowable RCS pressure to be 369 psia which is reduced to 275 psia to include allowances for instrument uncertainty, elevation head effects, reactor vessel to surge line ΔP, and analysis margin.

B. Heatup and Cooldown Rate (T.S. 2.1.2)

The RCS (except the pressurizer) temperature and pressure shall be limited in accordance with the limit lines shown on Figures 4.1 and 4.2 during heatup, cooldown, criticality, and inservice leak and hydrostatic testing.

2.3.2 Chemical and Volume Control System (T.S. 2.2)

A. Boric Acid Flow Paths - Shutdown (T.S. 2.2.1)

The flow path from the SIRWT to the RCS via a single HPSI pump shall only be established if:

2.3.2A 1) The RCS pressure boundary does not exist (i.e., mode 5 with the reactor vessel head removed or a 0.94 square inch or larger vent area per T.S. 2.1.6(4)), or

2) No charging pumps are operable with flow paths defined in T.S. 2.2.1a through 2.2.1c and the RCS heatup and cooldown rates shall be limited to those in Figures 4.1 and 4.2.

B. Charging Pumps - Shutdown (T.S. 2.2.3)

The flow path from the SIRWT to the RCS via a single HPSI pump shall be established only if:

1) The RCS pressure boundary does not exist (i.e., mode 5 with the reactor vessel head removed or a 0.94 square inch or larger vent area per T.S 2.1.6(4)), or

2) No charging pumps are operable and the RCS heatup and cooldown rates shall be limited to those in Figures 4.1 and 4.2.

2.3.3 Emergency Core Cooling System (T.S. 2.3)

A. Protection Against Low Temperature Overpressurization (T.S. 2.3(3))

The following limiting conditions shall be applied during scheduled heatups and cooldowns. Disabling of the HPSI pumps need not be required if the RCS is vented through at least a 0.94 square inch or larger vent, consistent with T.S. 2.1.6(4).

1) Whenever the reactor coolant system cold leg temperature is below 385°F, at least one (1) HPSI pump shall be disabled.

2) Whenever the reactor coolant system cold leg temperature is below 320°F, at least two (2) HPSI pumps shall be disabled.

3) Whenever the reactor coolant system cold leg temperature is below 270°F, all three (3) HPSI pumps shall be disabled.

In the event that no charging pumps are operable when the reactor coolant system cold leg temperature is below 270°F, a single HPSI pump may be made operable and utilized for boric acid injection to the core, with flow rate restricted to no greater than 120 gpm.

2.3.3A The normal procedure for starting the reactor is to first heat the reactor coolant to near operating temperature by running the reactor coolant pumps. The reactor is then made critical by withdrawing CEA's and diluting the boron concentration of the reactor coolant. With this mode of start-up, the energy stored in the reactor coolant during the approach to criticality is essentially equal to that during power operation and therefore all engineered safety features and auxiliary cooling systems are required to be fully operable.

B. The LTOP enable temperature is 385°F. Below this temperature the system is enabled while above it the high pressurizer pressure trip serves to actuate the PORVs.

2.4 Beltline Material Adjusted Reference Temperature (ART)

The calculation of the adjusted reference temperature (ART) for the reactor vessel beltline region has been performed using the NRC-accepted methodologies as described in Reference 3.2. Section 7.0, (Application of Surveillance Capsule Data to the Calculation of Adjusted Reference Temperature) was conservatively not used to refine the chemistry factor and the margin term. As noted in Section 2.3, a chemistry factor of 234.5°F and a fast neutron fluence of 1.501 x 10^{19} n/cm^2 was used for the limiting 3-410 welds. Use of Reference 3.8 has resulted in the determination of a revised best estimate value of 231.06°F, although not credited in these analyses.

The limiting ART values in the beltline region for FCS corresponding to the 20 Effective Full Power Years (EFPY) for the 1/4t and 3/4t locations (based on References 3.3, 3.15, and 3.16) are:

Location	ART	Material
1/4t	242.23° F	Welds 3-410 at 60° and 300°
3/4t	184.97° F	Welds 3-410 at 60° and 300°

The projected RT_{PTS} value for FCS which is currently calculated in accordance with 10 CFR 50.61 is 267° F (Reference 3.8 value refined from end of cycle value to license expiration on August 9, 2013) which corresponds to the 3-410 welds at 60° and 300° (i.e. 180° not limiting as previously noted in Section 2.1) for a tandem arc weld of weld wire heat numbers 27204/12008.

2.5 Pressure-Temperature Limits using limiting ART in the P-T curve calculation

2.5.1 Heatup and Cooldown Rate (T.S. 2.1.2)

The limits for T.S. 2.1.2 are presented in the subsection that follows. The analytical methods used to develop the RCS pressure-temperature limits for heatup and cooldown are based on NRC-accepted methodologies and discussed in Reference 3.2. The methodology is discussed below.

Before the radiation exposure of the reactor vessel exceeds the exposure for which they apply, Figures 4.1 and 4.2 shall be updated in accordance with the following criteria and procedures:

A. The curve in Figure 4.3 shall be used to predict the increase in transition temperature based on integrated fast neutron flux. If measurements on the irradiation specimens indicate a significant deviation from this curve, a new curve shall be constructed. If the embrittlement correlation (i.e. R.G. 1.99 or ASTM E900) changes, the figure shall be reviewed for assessment of the need of revision and revised if necessary.

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 shali be converted to an equivalent integrated fast neutron exposure (E>1 MeV). The predicted transition temperature shift to the end of the new period shall then be obtained from Figure 4.3.

C. The limit lines in Figures 4.1 and 4.2 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 NDTT of the reactor vessel flange and not subject to fast neutron flux. The lowest service temperature shall remain at 182°F because components related to this temperature are also not subject to fast neutron flux.

D. The requirements associated with Technical Specification 2.3(3) shall be reviewed and revised as necessary each time the curves of Figures 4.1 and 4.2 are revised.

2.5.1 The maximum allowable reactor coolant system pressure at any temperature is based upon the stress limitations for brittle fracture consideration. These limitations are derived by using the rules contained in Section III (and XI) of the ASME Code including Appendix G, Protection Against Nonductile Failure, and the rules contained in 10 CFR 50, Appendix G, Fracture Toughness Requirements, The ASME Code assumes that a crack 1-25/32 inches deep and 10-11/16 inches long exists (i.e. 1/4t location crack with an aspect ratio of 1:6) 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 nil-ductility 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 the transverse direction (RT_{NDT}) of -12°F.

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 and a conservative RT_{NDT} of 50°F has been established.

The initial RT_{NDT} value for the Fort Calhoun submerged arc vessel weldments was determined to be -56°F consistent with 10 CFR 50.61(2)(i) with a standard deviation of 17°F. The adjusted reference temperature (RT_{NDT}) was determined through the use of Regulatory Guide 1.99, Rev. 02 and 10CFR50.61 methods by summing the -56°F initial RT_{NDT} (10CFR50.61(2)(i)), the margin term of 66°F (10CFR50.61(2)(ii)) and the ΔRT_{NDT} (10CFR50.61(2)(iii)).

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>1 MeV) fluxes of the reactor vessel are described in Section 3.4.6 of the USAR.

2.5.1 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. 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 second removed irradiated reactor vessel surveillance specimen (Reference 3.11) combined with weld chemical composition data and reduced fluence core loading designs initiated in Cycle 8, indicated that the fluence at the end of 20.0 Effective Full Power Years (EFPY) at 1500 MWt would be 1.501×10^{9} n/cm^2 on the inside surface of the reactor vessel. Operation through fuel Cycle 19 will result in less than 20.0 EFPY.

The limit lines in Figures 4.1 and 4.2 are developed using the NRC approved ABB/CE methodology presented in Reference 3.2. References 3.15 and 3.16 represent the Fort Calhoun Station specific analyses. The general approach is summarized as follows:

a. Heatup and Cooldown Curves - In accordance with the ASME Code Section III Appendix G requirements, the general equations for determining the allowable pressure for any assumed rate of temperature change during Service Level A and B operation are:

$$2K_{IM} + K_{IT} < K_{IR}$$

where,

K_{IM} = Allowable pressure stress intensity factor, Ksi$\sqrt{\text{in}}$

K_{IT} = Thermal stress intensity factor, Ksi$\sqrt{\text{in}}$

K_{IR} = Reference stress intensity, Ksi$\sqrt{\text{in}}$

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.

2.5.1 The pressure-temperature limits provided in this report account for the temperature differential between the reactor vessel base metal and the reactor coolant bulk fluid temperature. Correction for elevation and RCS flow induced pressure differences between the reactor vessel beltline and pressurizer, are included in the development of the pressure-temperature limits. Consequently, the P-T limits are provided on coordinates of pressurizer pressure versus indicated RCS temperature.

2.5.1 The pressure correction factors are based upon the differential pressure due to the elevation difference between the reactor vessel wall adjacent to the bottom of the active core region, and the pressurizer pressure instrument nozzle. This term of the pressure correction factor is equal to 27.6 psi. The pressure correction factors are also based upon flow induced pressure drops across the reactor core through the hot leg pipe up to the surge line nozzle. This term of the pressure correction factor has two values which are dependent upon the Reactor Coolant Pump (RCP) combination utilized during operation. At temperatures of $T_c < 210°F$, the flow induced pressure drop is based upon the RCS flow rates resulting from two operating RCPs and is equal to 27.82 psia. At the temperatures of $T_c \geq 210°F$, the induced pressure drop is based upon the RCS flow rates resulting from three operating RCPs and is equal to 33.79 psi. Consequently, two pressure correction factors are utilized in correcting the reactor vessel beltline region pressure to pressurizer pressure depending upon the cold leg temperature. The following pressure correction factors have been utilized:

T_c (°F)	PRESSURE CORRECTION FACTOR (PSI)
≥210°F	61.5 psi
<210°F	55.5 psi

By explicitly accounting for the temperature differential between the flaw tip base metal temperature and the reactor coolant bulk fluid temperature, and the pressure differential between the beltline region of the reactor vessel and the pressurizer pressure measurement nozzle, the P-T limits are correctly represented on coordinates of pressurizer pressure and cold leg temperature. A temperature correction factor of 16°F has been utilized in the calculation of the P-T limits to account for temperature measurement uncertainties.

Pressure instrument loop uncertainties have not been included in the Pressure-Temperature limits since these uncertainties have been included in the setpoints of the Low Temperature Overpressure Protection (LTOP) system which has an enable temperature of 385°F. Including the pressure instrument loop uncertainties in the P-T limits therefore would have resulted in a redundant summation of these uncertainties which would be overly conservative.

b. Inservice Hydrostatic Test - The inservice hydrostatic test curve is developed in the same manner as in Section 2.5.1.a above with the exception that a safety factor of 1.5 is allowed by ASME III in lieu of 2.

2.5.1 c. Lowest Service Temperature = 50°F + 120°F + 12°F = 182°F. As indicated previously, an RT_{NDT} for all material with the exception of the reactor vessel beltline was established at 50°F. 10 CFR Part 50, Appendix G, IV.a.2 requires a lowest service temperature of RT_{NDT}+120°F for piping, pumps and valves. Additionally 12°F is added to account for insturment error. 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) - 56 = 569 psia, where 56 psi is the hydrostatic head correction factor.

d. Boltup Temperature = 10°F + 60°F + 12°F = 82°F. This temperature is conservatively based on the previous NDT methods witn the 60°F term representing an additional conservatism. At pressures below 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 previous NDTT methods. This temperature corresponds to the 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.

e. The temperature at which the heatup and cooldown rates change in Figures 4.1 and 4.2 reflects the setpoint at which the limiting heatup and cooldown rates with respect to the inlet temperature (T_c) change.

2.6 Minimum Temperature Requirements in the P-T curves

The minimum temperature requirements specified in Appendix G to 10 CFR 50 are applied to the P/T curves using the NRC-accepted methodologies as described in Reference 3.2.

The minimum temperature values applied to the P/T curves for FCS corresponding to 20 Effective Full Power Years (EFPY) as described above in Sections 2.5.1.d and 2.5.1.c, respectively are:

Location	Min Temperature
Boltup	82°F
Lowest Service Temperature	182°F

2.7 Surveillance Requirements

2.7.1 Reactor Coolant System and other components subject to ASME Section XI Boiler and Pressure Vessel Code Inspection and Testing Surveillance (T.S. 3.3).

The FCS post-irradiation surveillance capsule test program was established in accordance with ASTM-E185-66 and complies with 10CFR50 Appendix H. The test results are given in Reference 3.9 through 3.11. The test results do meet the credibility criteria of Regulatory Guide 1.99 Revision 2, with the exception of item A because the surveillance weld does not represent the controlling reactor vessel beltline weld material. The criteria are summarized below:

A. The surveillance program plate or weld duplicates the controlling reactor vessel beltline material in terms of ART:

B. Charpy data scatter does not cause ambiguity in the determination of the 30 ft-lb shift;

C. The measured shifts are consistent with the predicted shifts;

D. The capsule irradiation temperature is comparable to that of the vessel; and

E. Correlation monitor data are available and are consistent with the known data for that material.

The credible surveillance data have not been used to refine the chemistry factor and the margin term.

3. REFERENCES

3.1 NRC GL 96-03, "Relocation of Pressure-Temperature Limit Curves and Low Temperature Overpressure Protection System Limits". January 31, 1996.

3.2 CE NPSD-683, "Development of a RCS Pressure and Temperature Limits Report for the Removal of P-T Limits and LTOP Requirements from the Technical Specifications, CEOG Task 942". (Latest Approved Revision)

3.3 FCS Technical Specifications, Amendment No. 161.

3.4 WCAP-14040-NP-A, Revision 1 (Section 2.2 Neutron Fluence Calculation), "Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves", January 1996.

3.5 Draft Regulatory Guide DG-1053, "Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence", June 1996.

3.6 Draft Regulatory Guide DG-1025, "Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence", September 1993.

3.7 Letter from OPPD (S. K. Gambhir) to NRC (Document Control Desk) dated January 30, 1998, (LIC-98-0009) containing as an attachment, SE-REA-95-003, "Fast Neutron Fluence Evaluations for Fort Calhoun Unit 1 Reactor Pressure Vessel", November 1995.

3.8 Letter from OPPD (S.K. Gambhir) to NRC (Document Control Desk), dated October 13, 1997 (LIC-97-0159).

3.9 FCS USAR Table 4.5-4, "Capsule Removal Schedule"

3.10 Letter from OPPD (W.C. Jones) to NRC (H.R. Denton), dated January 23, 1981 (LIC-81-0011). Enclosure: TR-O-MCM-001, Rev. 01, "Evaluation of Irradiated Capsule W-225".

3.11 Letter from OPPD (W.C. Jones) to NRC (D.G. Eisenhut), dated April 25, 1984. Enclosure; TR-O-MCM-002, "Post-Irradiation Evaluation of Reactor Vessel Surveillance Capsule W-265".

3.12 Letter from OPPD (T.L. Patterson) to NRC (Document Control Desk), dated December 9, 1994. Enclosure: BAW-2226 "Evaluation of Irradiated Capsule W-275".

3.13 TR-O-MCD-001, "OPPD FCS Unit No. 1 Evaluation of Baseline Specimens Reactor Vessel Materials Surveillance Program", March 22, 1977.

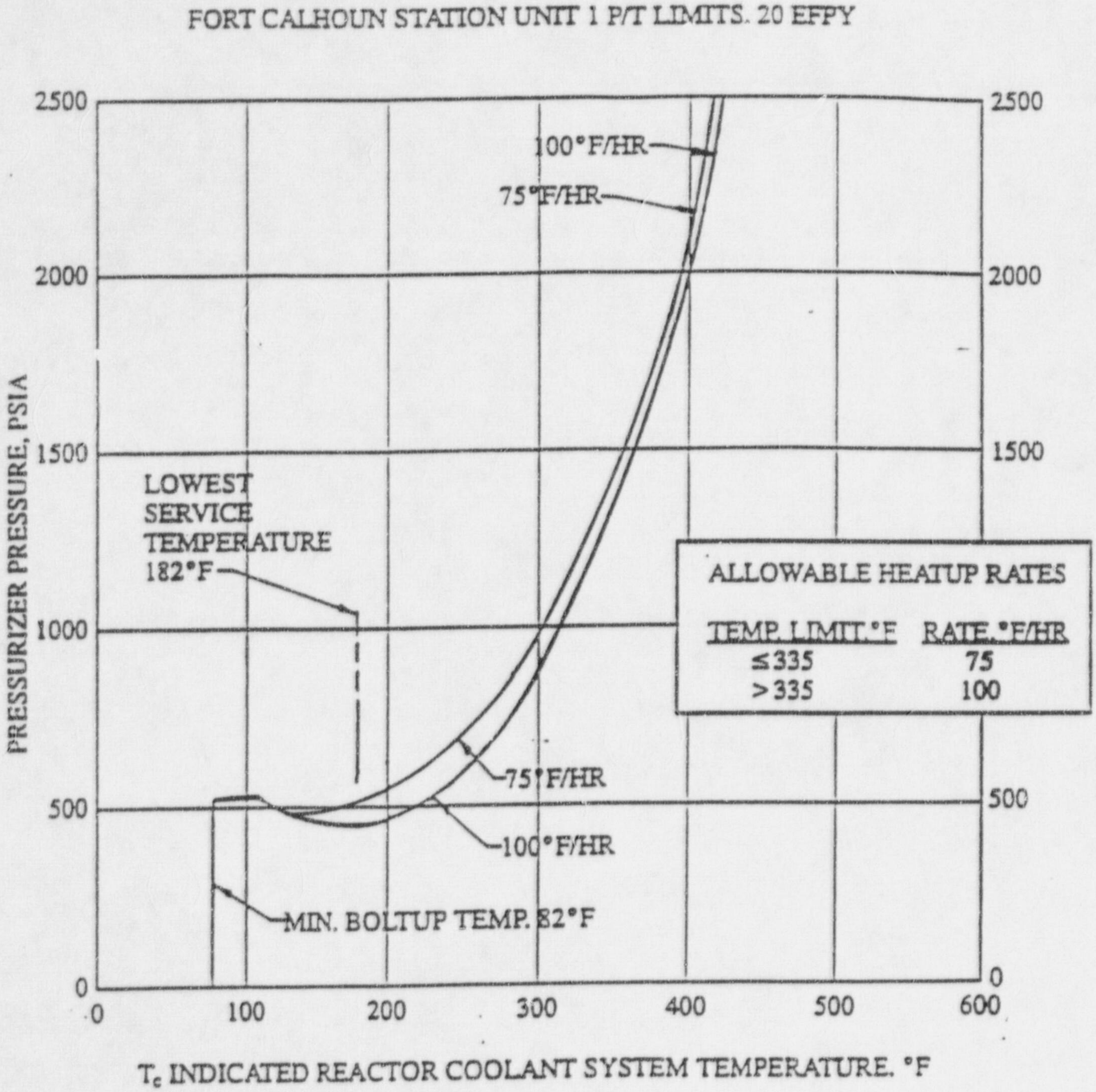
3.14 O-PENG-ER-012 Rev. 01, "OPPD Interim Report on ASME Appendix G Evaluation of Step Changes in RCS Temperatures (CEOG Task 1004)", February 1998.

3.15 Letter O-MPS-90-043 from ABB/CE (A.A. Ostrov) to OPPD (K.C. Holthaus), dated June 14, 1990.

3.16 Letter O-MPS-90-053 from ABB/CE (A.A. Ostrov) to OPPD (K.C. Holthaus), dated July 20, 1990.

4. OPERATING FIGURES

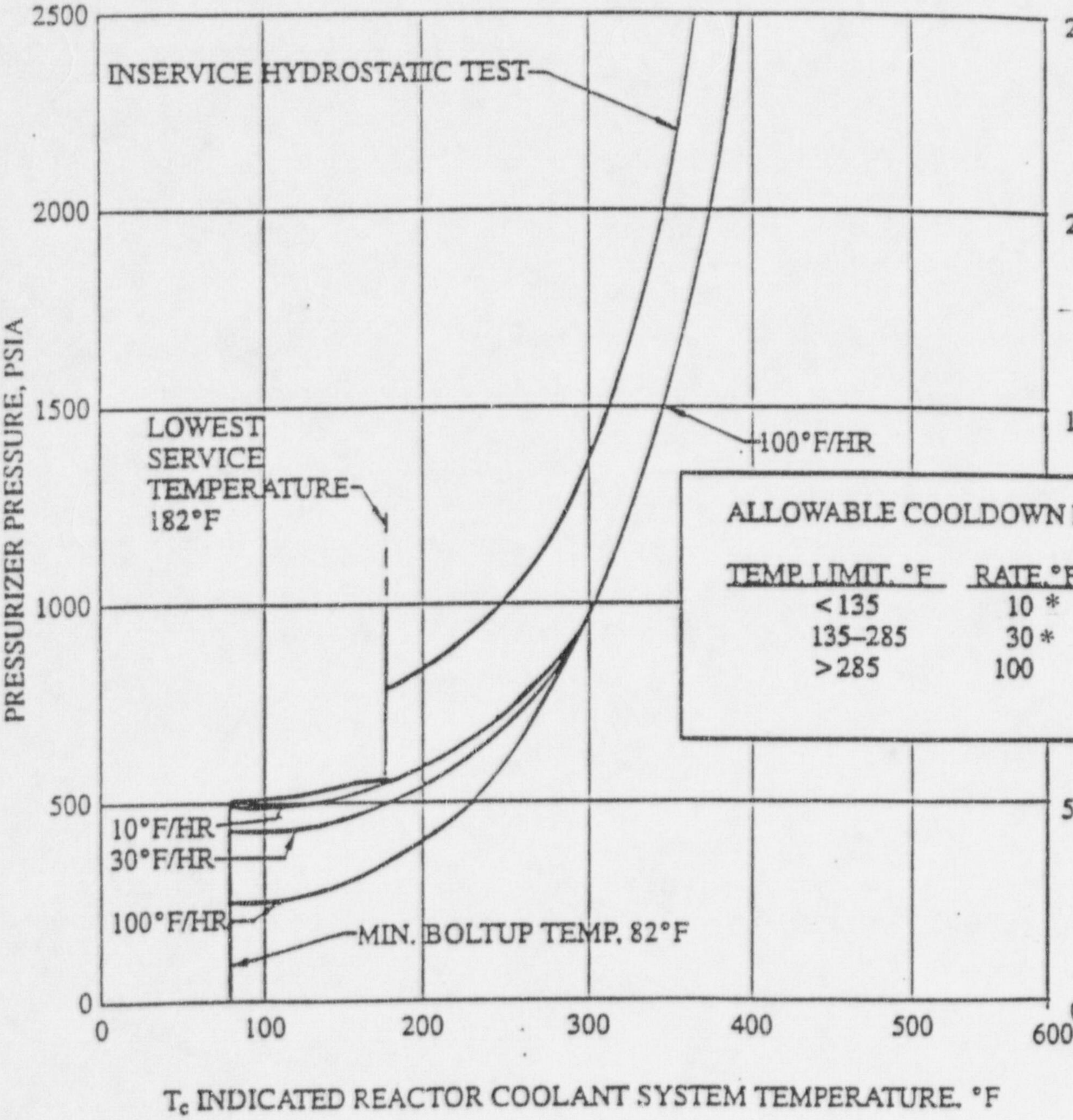
The referenced operating Figures follow.



Note: Startup of the first RCP is a normal plant evolution and no heatup or cooldown limit violations will occur as analyzed in Reference 3.14, which demonstrates compliance with the criteria of 10 CFR 50 Appendix G and ASME Section III Appendix G.

RCS Pressure–Temperature Limits for Heatup	Omaha Public Power District Fort Calhoun Station– Unit No. 1	Figure 4.1

FORT CALHOUN STATION UNIT 1 P/T LIMITS, 20 EFPY
COOLDOWN AND INSERVICE TEST

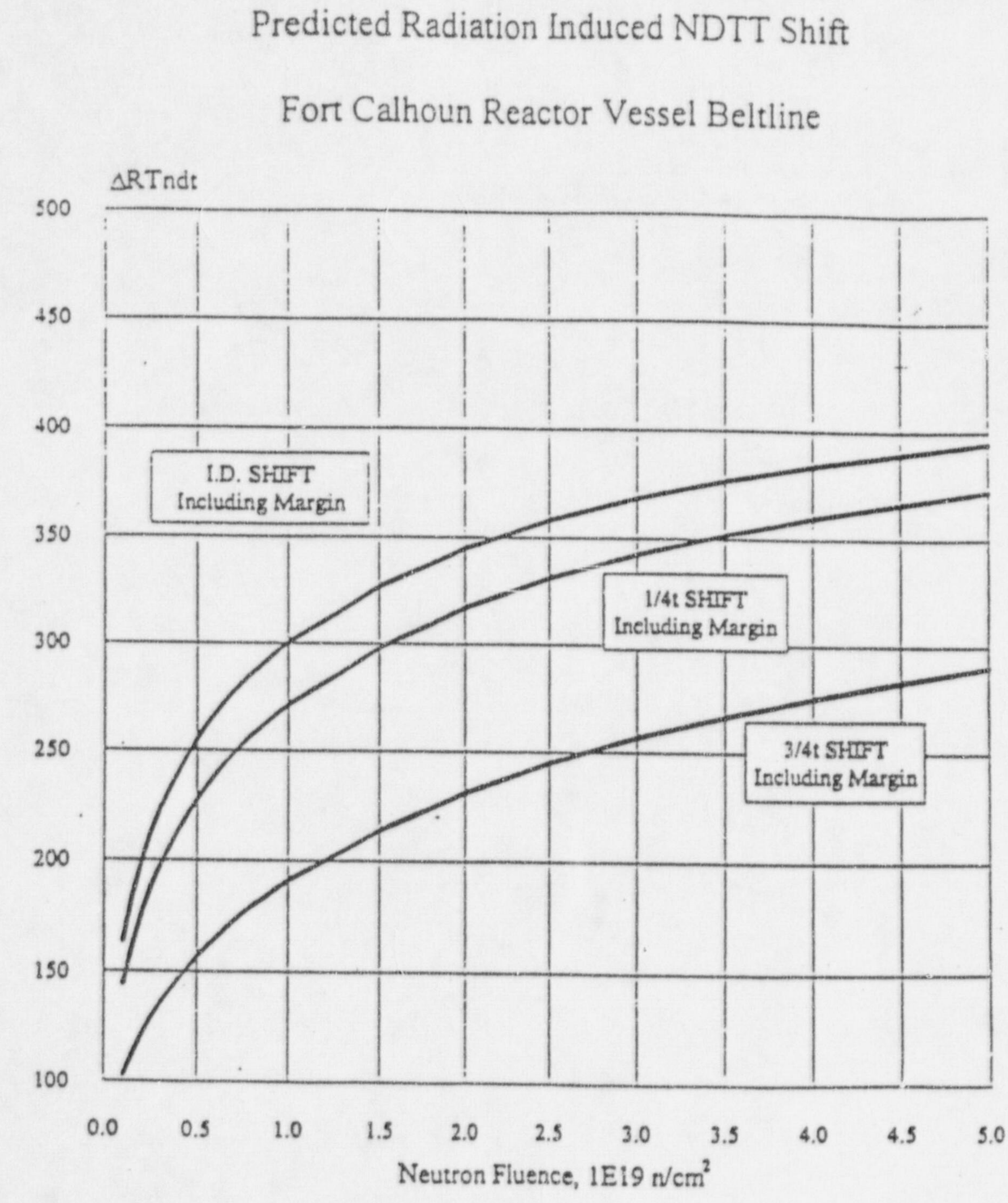


ALLOWABLE COOLDOWN RATE

TEMP. LIMIT, °F	RATE, °F/HR
<135	10 *
135–285	30 *
>285	100

* Startup of the first RCP is a normal plant evolution and no heatup or cooldown limit violations will occur as analyzed in Reference 3.14, which demonstrates compliance with the criteria of 10 CFR 50 Appendix G and ASME Section III Appendix G.

RCS Pressure–Temperature Limits for Cooldown	Omaha Public Power District Fort Calhoun Station– Unit No. 1	Figure 4.2



Predicted Radiation Induced NDTT Shift	Omaha Public Power District Fort Calhoun Station-Unit No. 1	Figure 4.3