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

(a) The curve in Figure 2-3 shall be used to predict the increase in transition temperature based on integrated fast neutron flux. If measurements on the irradiation specimens indicate a deviation from this curve, a new curve shall be constructed.

(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≥1 MeV). For this plant, based upon surveillance materials tests, weld chemical composition data, and the effect of a reduced vessel fluence rate provided by core load designs beginning with fuel Cycle 8, the predicted surface fluence at the initial reactor vessel beltline weld material for 40 years at 1500 MWt and an 80% load factor is 2.9 x 10^{19} n/cm^2. The flux reduction applied 7
255 to the fluence calculations was based on a Cycle 1-9 average
and cycle 8 average azimuthal flux distribution plots generated using DOT 4.3. The predicted transition temperature shift to the end of the new period shall then be obtained from Figure 2-3.

(c) The limit lines in Figures 2-1A 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 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 Technical Specification 2.3(3) shall be revised each time the curves of Figures 2-1A 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 a rate of 100°F in any one hour period and for 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 III[2] of the ASME Code including Appendix G, Protection Against Non-ductile Failure, 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 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 transverse direction (RT_{NDT}) of -12°F.

The mean 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 the proposed 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>1 MeV) fluxes of the reactor vessel are described in Section 3.4.6 of the USAR, except that the integrated
2.55 fast neutron flux (E>1 MeV) is 2.9x10^{19} n/cm^2, including tolerance at the inside surface of the critical reactor vessel beltline weld material, over the 40 year design life of the vessel.[5]

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≥1 MeV) exposure of the reactor vessel at the critical reactor vessel beltline location including tolerance is computed to be
2.55 2.9x10^{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 80% load factor. The predicted shift at this location at the 1/4t depth from the inner surface is 332°F, including margin, and was calculated using the shift prediction equation of the proposed 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 second removed irradiated reactor vessel surveillance specimen(8), combined with weld chemical composition data and reduced core loading designs initiated in Cycle 8, indicates that 14 the fluence at the end of 15.0 Effective Full Power Years (EFPY) at 1500 MWt will be 1.4×10^{19} n/cm² on the inside surface of the reactor vessel. This results in a total shift of the RT_{NDT} of 285°F, including margin, for the area of greatest sensitivity (weld metal) at the 1/4t location as determined from Figure 2-3. Operation through fuel Cycle 16 15 will result in less than 15.0 14 EFPY.

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.

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

K_{IR} = Allowance stress intensity factor at temperatures 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.

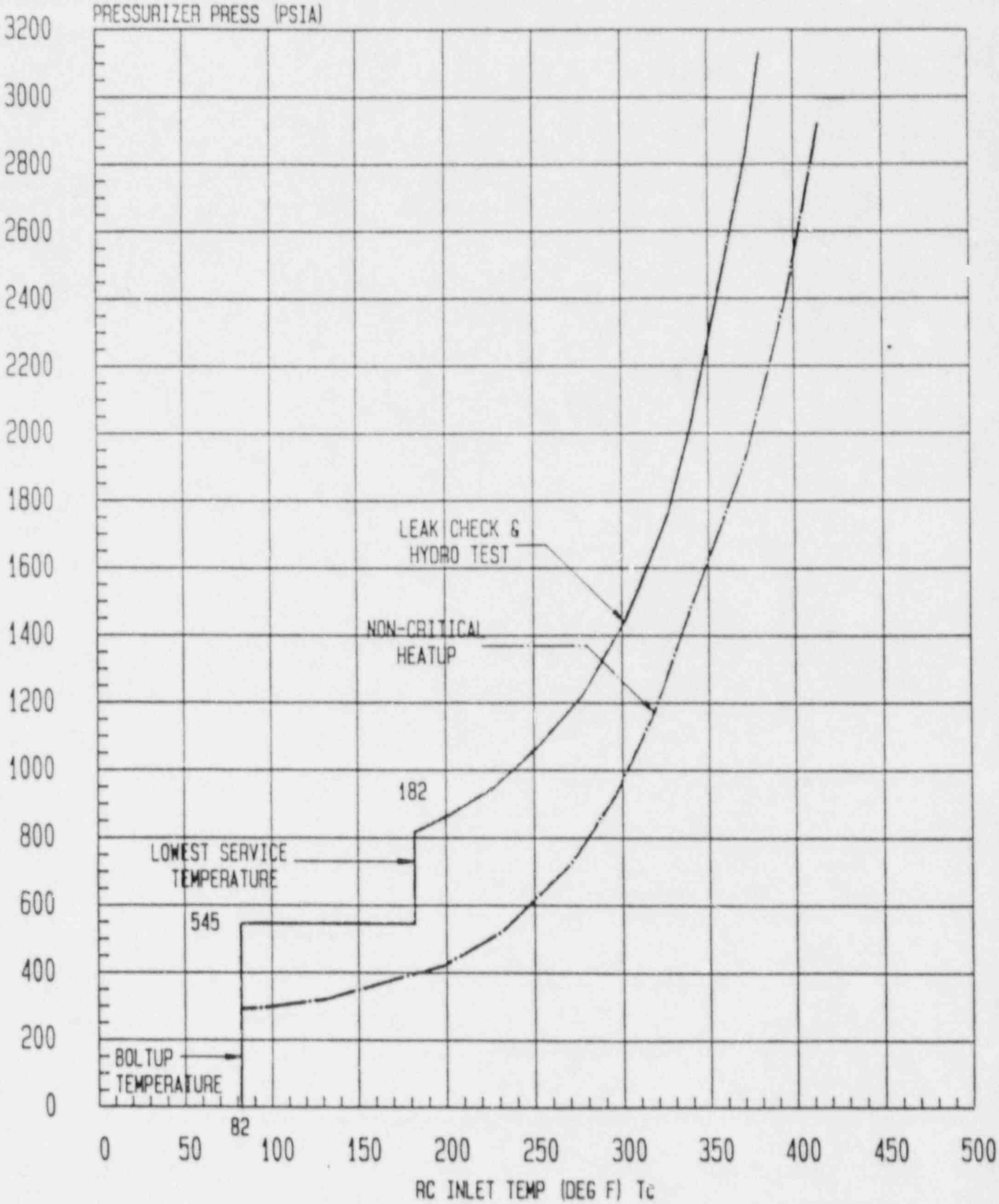
K_{IT} = Stress intensity factor radial thermal gradient.

The above equation is applied to the reactor vessel beltline. For plant heatup the thermal stress is opposite in sign from the pressure stress and consideration of a heatup rate would allow for a higher pressure. For heatup it is therefore conservative to consider an isothermal heatup or K_{IT} = 0.

For plant cooldown thermal and pressure stress are additive.

RCS PRESS-TEMP LIMITS HEATUP 14 EFPY

REACTOR NOT CRITICAL 1500 MWt

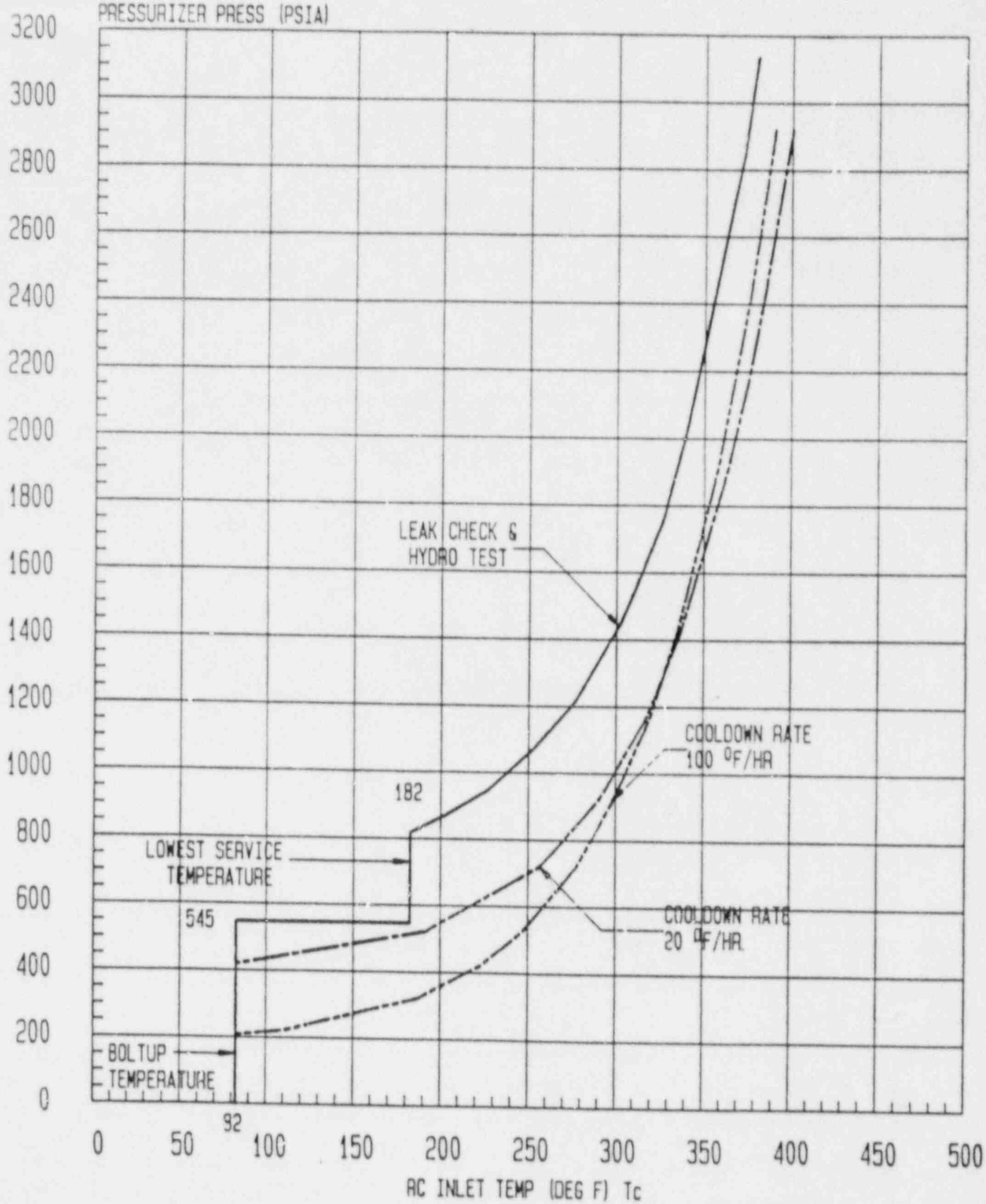


FORT CALHOUN
TECHNICAL SPECIFICATIONS

FIGURE
2-1A

RCS PRESS-TEMP LIMITS COOLDOWN 14 EFPY

REACTOR NOT CRITICAL 1500 MWt

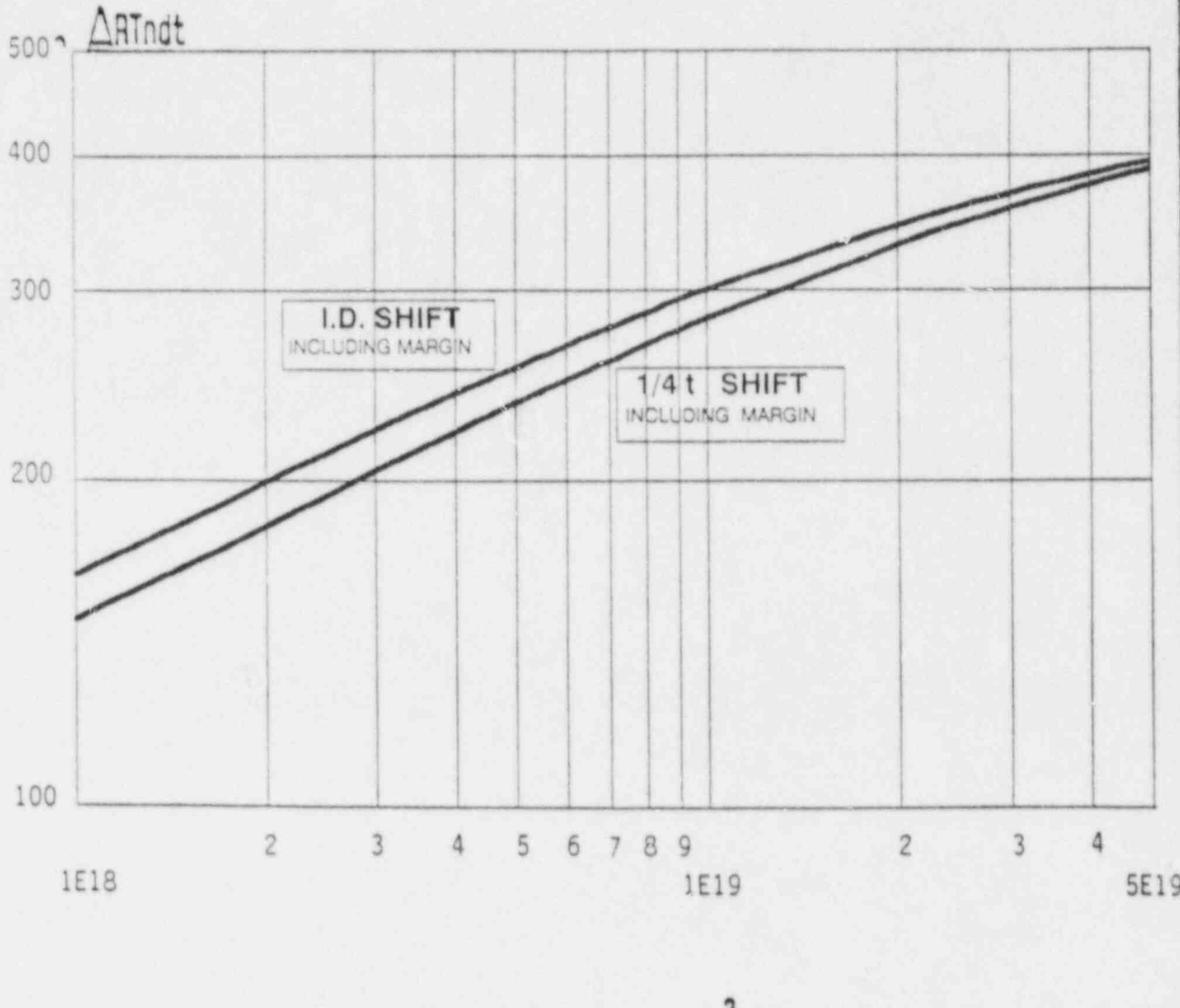


FORT CALHOUN
TECHNICAL SPECIFICATIONS

FIGURE
2-1B

PREDICTED RADIATION INDUCED NDTT SHIFT

FORT CALHOUN REACTOR VESSEL BELTLINE



ATTACHMENT B

JUSTIFICATION, DISCUSSION

NO SIGNIFICANT HAZARDS CONSIDERATIONS

Documentation of the chemical content of all Fort Calhoun reactor vessel beltline materials was completed in 1986 (Reference 2). The lower shell longitudinal weld seams (3-410), which consist of three different weld wire heats, were found to be limiting as a result of high copper and nickel content. Since the combination of these welds is unknown, the most limiting weld is used in the RT_{NDT} analysis. Weld wire heat 12008, flux lot 3774, containing 0.23 w/o Cu and 0.95 w/o Ni was found to be limiting when using the shift prediction equation of 10 CFR 50.61. This weld was also assumed for use with Reg. Guide 1.99, Draft Rev. 2, when preparing the facility license change to update the heatup and cooldown limit curves to 15.0 EFPY (Reference 3). Further investigation revealed that weld wire heat 27204, flux lot 3774, with 0.22 w/o Cu and 1.02 w/o Ni is more limiting when using the Draft Reg. Guide methodology. Using the chemistry factor associated with this weld wire heat in the Reg. Guide 1.99, Draft Rev. 2 equation demonstrates that the existing Technical Specification heatup and cooldown limit curves (Reference 4) are non-conservative.

The valid lifetime of the existing 15.0 EFPY curves has been reanalyzed using the fluence prediction equation developed in Reference 5 and the more limiting chemistry factor associated with the 3-410 weld seams. The Reference 5 fluence prediction equation for the longitudinal 3-410 weld is:

$$\varnothing = (8.8\times10^{18})(0.68) + \frac{(EFPY - 5.92)(4.8\times10^{19})}{32}(0.50)\ n/cm^2$$

This equation takes additional credit for the vessel flux reduction program which was previously not taken in the Reference 3 submittal. The results of this analysis indicate that the current heatup and cooldown limit curves are valid to only 14.0 EFPY. Since Fort Calhoun Station has been operating for less than 10 EFPY, no challenge to the reactor coolant system or violation of Technical Specifications has occurred from using the existing curves.

This proposed amendment corrects the labels of the heatup and cooldown curves (Figures 2-1A and 2-1B) from 15.0 to 14.0 EFPY. Figure 2.3 is corrected to reflect the more conservative shift prediction equation associated with the limiting weld wire heat in the lower longitudinal weld seams. The predicted 40 year integrated flux was also revised to be consistent with the fluence prediction equation used in this assessment. Reference to 15.0 EFPY in the Basis Section is revised to 14.0 EFPY and the corresponding change from Cycle 16 to Cycle 15 is also made.

Significant Hazards Considerations

The proposed amendment to the Technical Specification does not involve an unreviewed safety question because the operation of the Fort Calhoun Station in accordance with this change would not:

(1) Increase the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the safety analysis report. The proposed revision to the Technical Specification heatup and cooldown limit curves imposes more conservative limits on operation by revising the valid operating life of the existing curves from 15 EFPY to 14 EFPY. There has been no challenge to the reactor coolant system associated with using the previous curves since the Fort Calhoun Station has currently been operating for less than 10 EFPY. Therefore this amendment would not increase the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the safety analysis report.

(2) Create the possibility for an accident or malfunction of a different type than any evaluated previously in the safety analysis report. This amendment only revises the label defining the lifetime in EFPY of the Technical Specification heatup and cooldown limit curves. These curves are bounded by the existing Safety Analysis Report. There are no anticipated changes to the current operating practices. Therefore, the possibility of an accident or malfunction of a different type than any evaluated previously in the safety analysis report would not be created.

(3) Reduce the margin of safety as defined in the basis for any Technical Specification. The revised operating life for the existing curves was determined using a more conservative chemistry factor along with the shift prediction equation, including the appropriate 2σ margin, as presented in Regulatory Guide 1.99, Draft Rev. 2. Therefore, the margin of safety as defined in the basis for any Technical Specification is not reduced.

Based on the above considerations, OPPD does not believe that this amendment involves a significant hazards consideration.

References

(1) Docket No. 50-285

(2) Letter from OPPD (R. L. Andrews) to NRC (A. C. Thadani), Fort Calhoun Specific Weld Chemistry Data Reporting Requirements of 10 CFR 50.61, dated January 23, 1986 (LIC-86-024)

(3) Letter from OPPD (R. L. Andrews) to NRC (A. C. Thadani), Revision of Application for Amendment of Heatup and Cooldown Limit Curves, dated July 10, 1986 (LIC-86-309)

(4) Amendment No. 100 to Facility Operating License DPR-40, September 8, 1986

(5) Letter from OPPD (R. L. Andrews) to NRC, Revised Pressurized Thermal Shock Analysis, dated December 21, 1987 (LIC-87-692)