UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

OMAHA PUBLIC POWER DISTRICT

DOCKET NO. 50-285

FORT CALHOUN STATION, UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 114
License No. DPR-40

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by the Omaha Public Power District (the licensee) dated February 19, 1988, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, as amended, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance: (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this license amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, Facility Operating License No. DPR-40 is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 3.B. of Facility Operating License No. DPR-40 is hereby amended to read as follows:

   B. Technical Specifications

   The Technical Specifications contained in Appendix A as revised through Amendment No. 114, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. The license amendment is effective as of its date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

José A. Calvo

Jose A. Calvo, Director
Project Directorate - IV
Division of Reactor Projects - III,
IV, V and Special Projects
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: June 28, 1988

ATTACHMENT TO LICENSE AMENDMENT NO. 114

FACILITY OPERATING LICENSE NO. DPR-40

DOCKET NO. 50-285

Revise Appendix "A" Technical Specifications as indicated below. The revised pages are identified by amendment number and contain vertical lines indicating the area of change.

Remove Pages	Insert Pages
2-4	2-4
2-5	2-5
2-6	2-6
Figure 2-1A	Figure 2-1A
Figure 2-1B	Figure 2-1B
Figure 2-3	Figure 2-3

(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 2.55×10^{19} n/cm^2. The flux reduction applied to the fluence calculations was based on Cycle 1-7 average and Cycle 8 average azimuthal flux distribution plots generated used 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 boilup 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.

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 laterial 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 ($E \geq 1$ MeV) fluxes of the reactor vessel are described in Section 3.4.6 of the USAR, except that the integrated fast neutron flux ($E \geq 1$ MeV) is 2.55×10^{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.[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 ($E \geq 1$ MeV) exposure of the reactor vessel at the critical reactor vessel beltline location including tolerance is computed to be 2.55×10^{19} n/cm^2 at the vessel inside surface for 40 years operation at

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, indicated that the fluence at the end of 14.0 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 15 will result in less than 14.0 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.

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

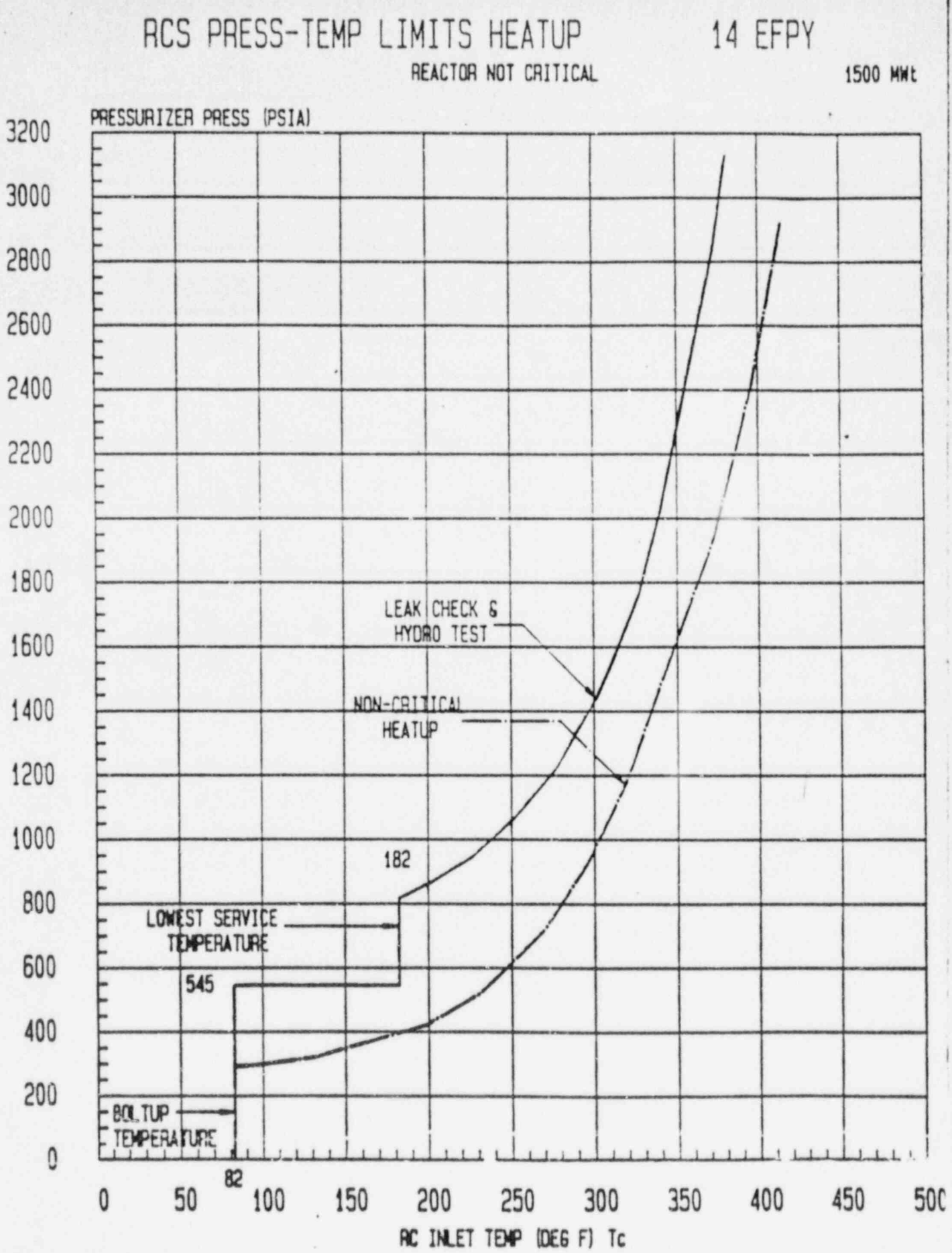
K_{IR} = Allowance stress intensity factor at temperature related to RT_{NDT} (ASME III Figure G-2110.1).

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 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.



FORT CALHOUN
TECHNICAL SPECIFICATIONS

Amendment No. 78,77,100, 114

FIGURE
2-1A

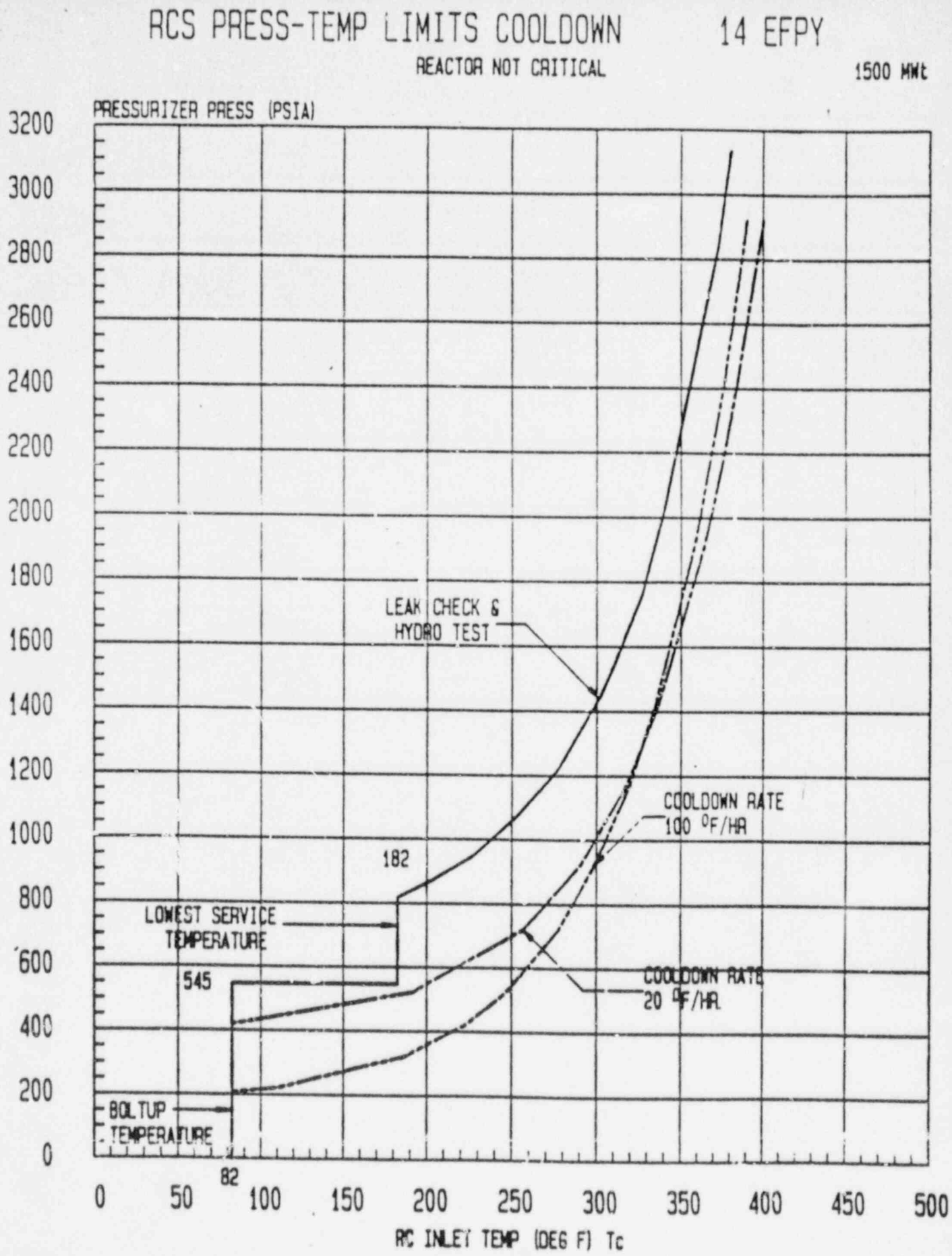


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FIGURE
2-1B

PREDICTED RADIATION INDUCED NDTT SHIFT

FORT CALHOUN REACTOR VESSEL BELTLINE

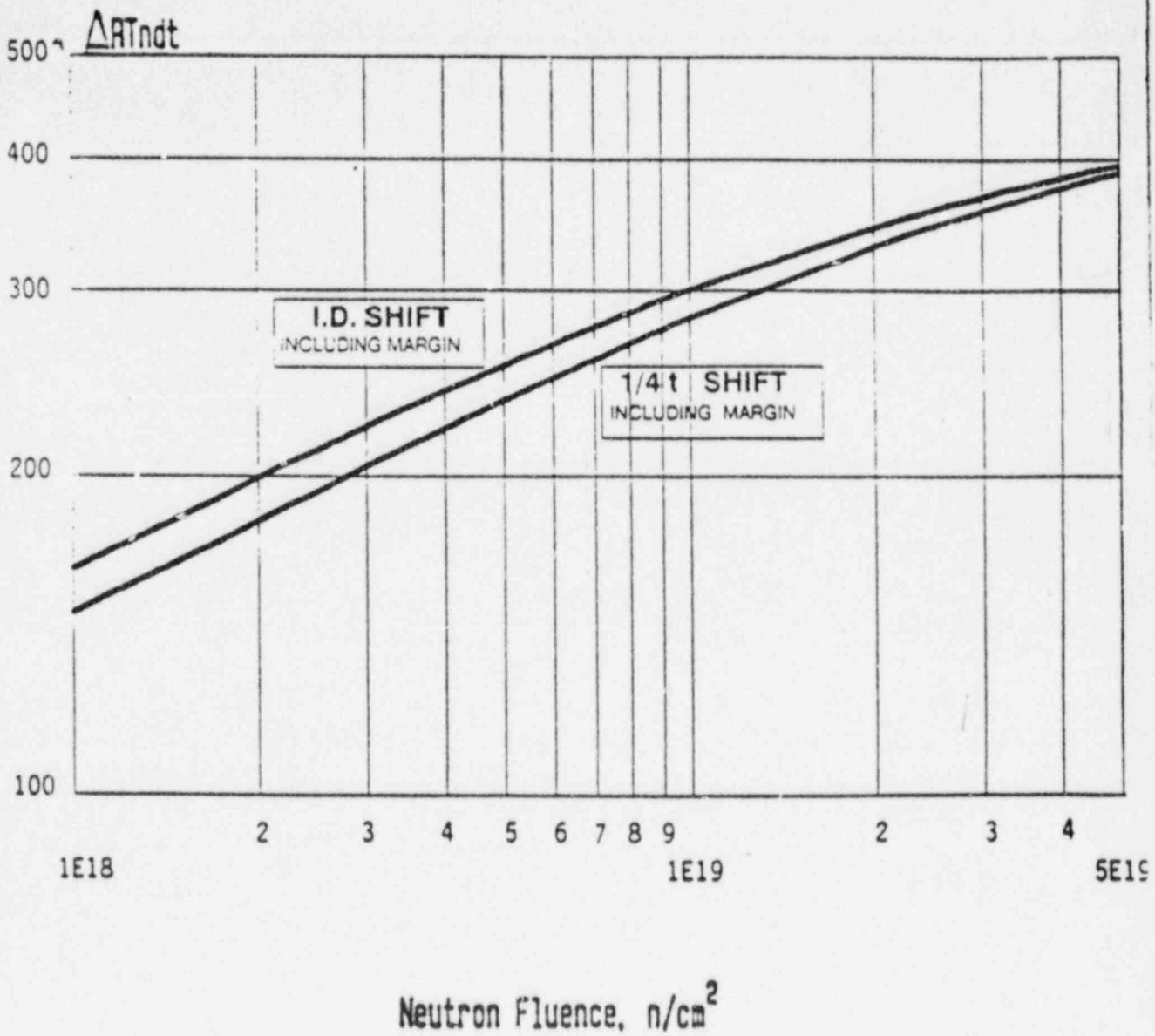

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FIGURE
2-3

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