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. 100
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 April 25, 1986 as revised by letter dated July 10, 1986, 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, 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 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.100 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of the date of its issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

A.C. Thadani

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PWR Project Directorate #8
Division of PWR Licensing-B

Attachment:
Changes to the Technical
Specifications

Date of Issuance: September 8, 1986

ATTACHMENT TO LICENSE AMENDMENT NO. 100

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
2-7	2-7
2-7a	2-7a
2-22	2-22
2-23a	2-23a
Figure 2-1A	Figure 2-1A
Figure 2-1B	Figure 2-1B
Figure 2-3	Figure 2-3

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 ($E \geq 1$ 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 2.9×10^{19} n/cm^2. The flux reduction applied to the fluence calculations was based on a Cycle 1-9 average azimuthal flux distribution plot 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 fast neutron flux (E>1 MeV) is 2.9×10^{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 2.9×10^{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 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 the fluence at the end of 15.0 Effective Full Power Years (EFPY) at 1500 MWt will be $1.4x10^{19}$ n/cm^2 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 will result in less than 15.0 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 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.

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

M_M = ASME III, Figure G-2214-1

P = Pressure, psia

R = Vessel Radius - in.

t = Vessel Wall Thickness - in.

$$K_{IT} = M_T \Delta T_W$$

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 zero for heatup.
$\frac{M_M}{t}$ R is also a constant = B.
Therefore:

$K_{IR} = AP + B$

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

K_{IR} is then varied as a function of temperature from Figure G-2110-1 of ASME III and the allowable pressure calculated. Hydrostatic head (48 psi) and instrumentation errors (12°F and 32 psi) are considered when plotting the curves.

B. System Hydrostatic Test - The system 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 + 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. Below this temperature a pressure of 20 percent of the system hydrostatic test pressure (.20)(3125) - 48 - 32 psi = 545 psia cannot be exceeded.

D. Boltup Temperature = 10°F + 60°F + 12°F = 82°F. At pressure below 545 psia, a minimum vessel temperature must be maintained to comply with the manufacturer's specifications for tensioning the vessel head.

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

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.

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) Omaha Public Power District, Fort Calhoun Station Unit No. 1, Evaluation of Irradiated Capsule W-225, Revision 1, August 1980.

(6) Technical Specification 2.3(3)

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

(8) Omaha Public Power District, Fort Calhoun Station Unit No. 1, Evaluation of Irradiated Capsule W-265, March 1984

2.0 LIMITING CONDITIONS FOR OPERATION
2.3 Emergency Core Cooling System (Continued)

(3) Protection Against Low Temperature Overpressurization

The following limiting conditions shall be applied during scheduled heatups and cooldowns. Disabling of the HPSI pumps need not be required if the reactor vessel head, a pressurizer safety valve, or a PORV is removed.

Whenever the reactor coolant system cold leg temperature is below 320°F, at least one (1) HPSI pump shall be disabled.

Whenever the reactor coolant system cold leg temperature is below 312°F, at least two (2) HPSI pumps shall be disabled.

Whenever the reactor coolant system cold leg temperature is below 271°F, all three (3) HPSI pumps shall be disabled.

In the event that no charging pumps are operable, a single HPSI pump may be made operable and utilized for boric acid injection to the core.

Basis

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 boron in the reactor coolant. With this mode of start-up, the energy stored in the reactor coolant during the approach to criticality is substantially equal to that during power operation and therefore all engineered safety features and auxiliary cooling systems are required to be fully operable. During low power physics tests at low temperatures, there is a negligible amount of stored energy in the reactor coolant; therefore, an accident comparable in severity to the design basis accident is not possible and the engineered safeguards systems are not required.

The SIRW tank contains a minimum of 283,000 gallons of usable water containing at least 1700 ppm boron[1]. This is sufficient boron concentration to provide a shutdown margin of 5%, including allowances for uncertainties, with all control rods withdrawn and a new core at a temperature of 60°F.[2]

The limits for the safety injection tank pressure and volume assure the required amount of water injection during an accident and are based on values used for the accident analyses. The minimum 116.2 inch level corresponds to a volume of 825 ft^3 and the maximum 128.1 inch level corresponds to a volume of 895.5 ft^3.

Prior to the time the reactor is brought critical, the valving of the safety injection system must be checked for correct alignment and appropriate valves locked. Since the system is used for shutdown cooling, the valving will be changed and must be properly aligned prior to start-up of the reactor.

be available for emergency core cooling, but the contents of one of the tanks is assumed to be lost through the reactor coolant system. In addition, of the three high-pressure safety injection pumps and the two low-pressure safety injection pumps, for large break analysis it is assumed that two high pressure and one low pressure operate while only one of each type is assumed to operate in the small break analysis[5]; and also that 25% of their combined discharge rate is lost from the reactor coolant system out of the break. The transient hot spot fuel clad temperatures for the break sizes considered are shown on FSAR, Appendix K, Tables 1-19 (Amendment No. 34).

Inadvertent actuation of three (3) HPSI pumps and three (3) charging pumps, coincident with the opening of one of the two PORV's, would result in a peak primary system pressure of 1190 psia. 1190 psia corresponds with a minimum permissible temperature of 320°F on Figure 2-1B. Thus, at least one HPSI pump is disabled at 320°F.

Inadvertent actuation of two (2) HPSI pumps and three (3) charging pumps, coincident with the opening of one of the two PORV's, would result in a peak primary system pressure of 1040 psia. 1040 psia corresponds with a minimum permissible temperature of 312°F on Figure 2-1B. Thus, at least two HPSI pumps will be disabled at 312°F.

Inadvertent actuation of one (1) HPSI and three (3) charging pumps, coincident with opening of one of the two PORV's, would result in a peak primary system pressure of 685 psia. 685 psia corresponds with a minimum allowable temperature of 271°F on Figure 2-1B. Thus, all three HPSI pumps will be disabled at 271°F.

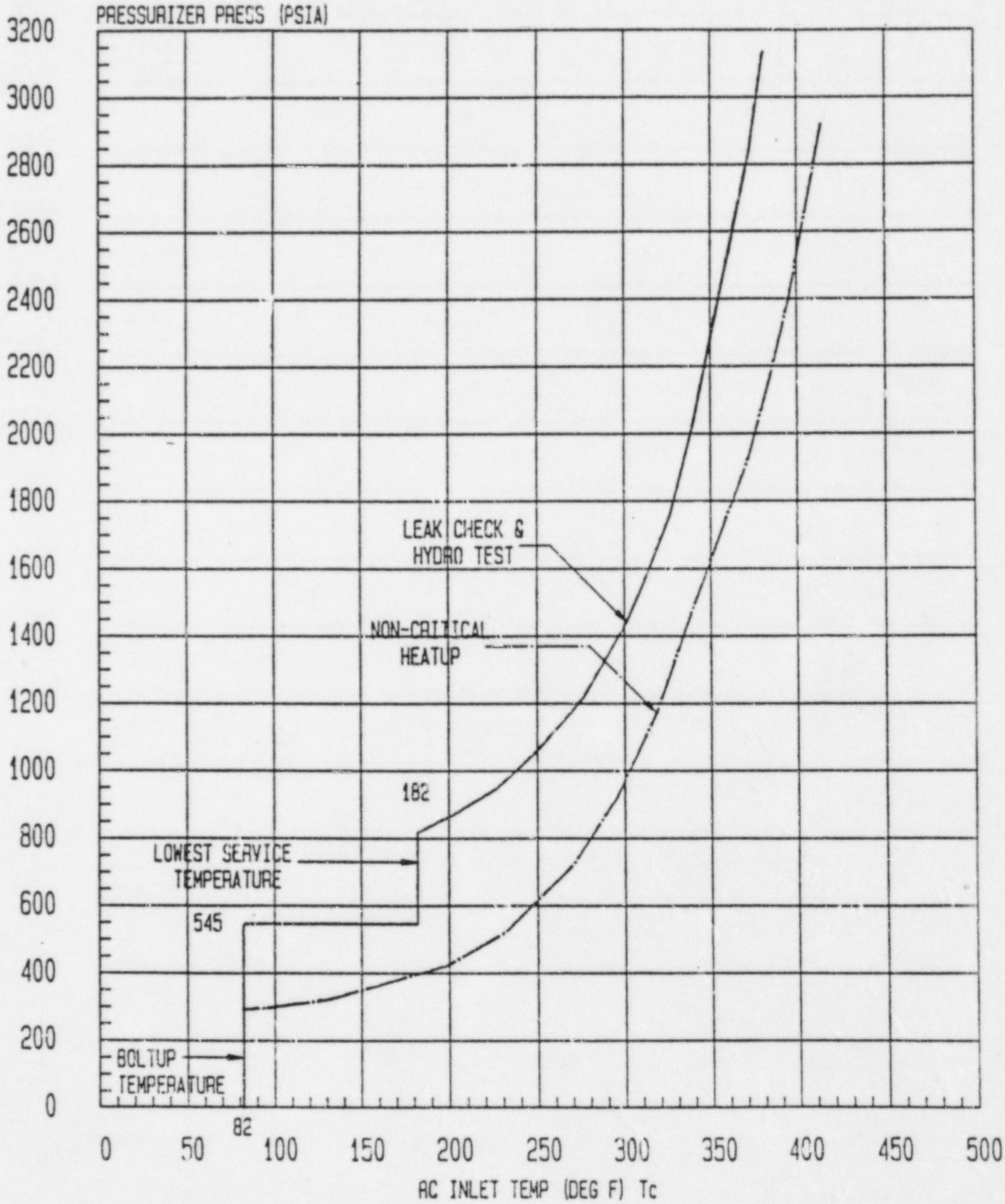
Inadvertent actuation of three (3) charging pumps, coincident with the opening of one of the two PORV's, would result in a peak primary system pressure of 160 psia. 160 psia would correspond with a minimum allowable temperature that is less than the 82°F boltup temperature limit on Figure 2-1B. Therefore, operation of the charging pumps need not be restricted.

Removal of the reactor vessel head, one pressurizer safety valve, or one PORV provides sufficient expansion volume to limit any of the design basis pressure transients. Thus, no additional relief capacity is required.

Technical Specification 2.2(1) specifies that, when fuel is in the reactor, at least one flow path shall be provided for boric acid injection to the core. Should boric acid injection become necessary, and no charging pumps are operable, operation of a single HPSI pump would provide the required flow path.

RCS PRESS-TEMP LIMITS HEATUP 15 EFPY

REACTOR NOT CRITICAL 1500 MWt



FORT CALHOUN
TECHNICAL SPECIFICATIONS

FIGURE
2-1A

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RCS PRESS-TEMP LIMITS COOLDOWN 15 EFPY

REACTOR NOT CRITICAL

1500 MWt

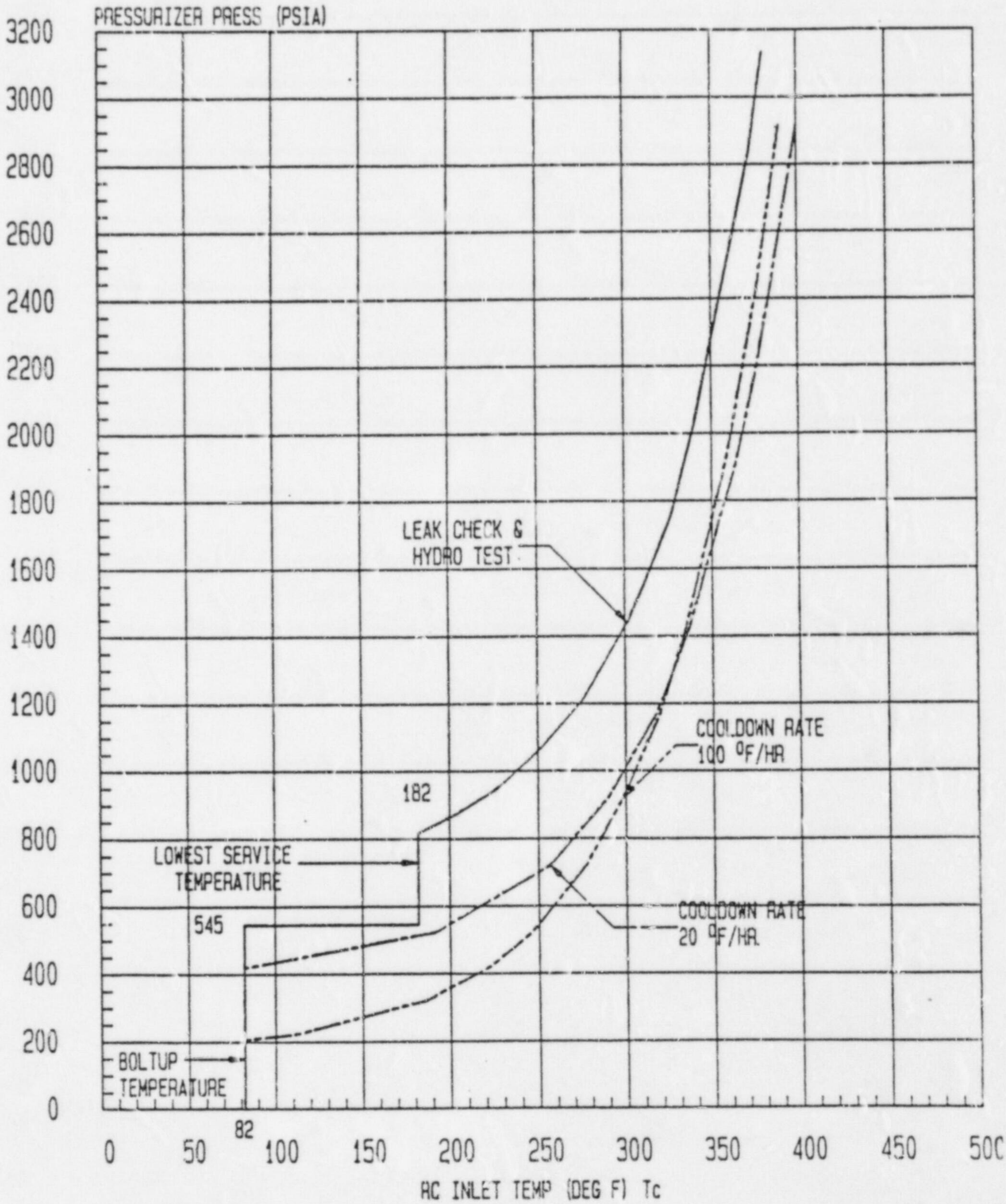

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FIGURE
2-1B

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PREDICTED RADIATION INDUCED NDTT SHIFT

FORT CALHOUN REACTOR VESSEL BELTLINE

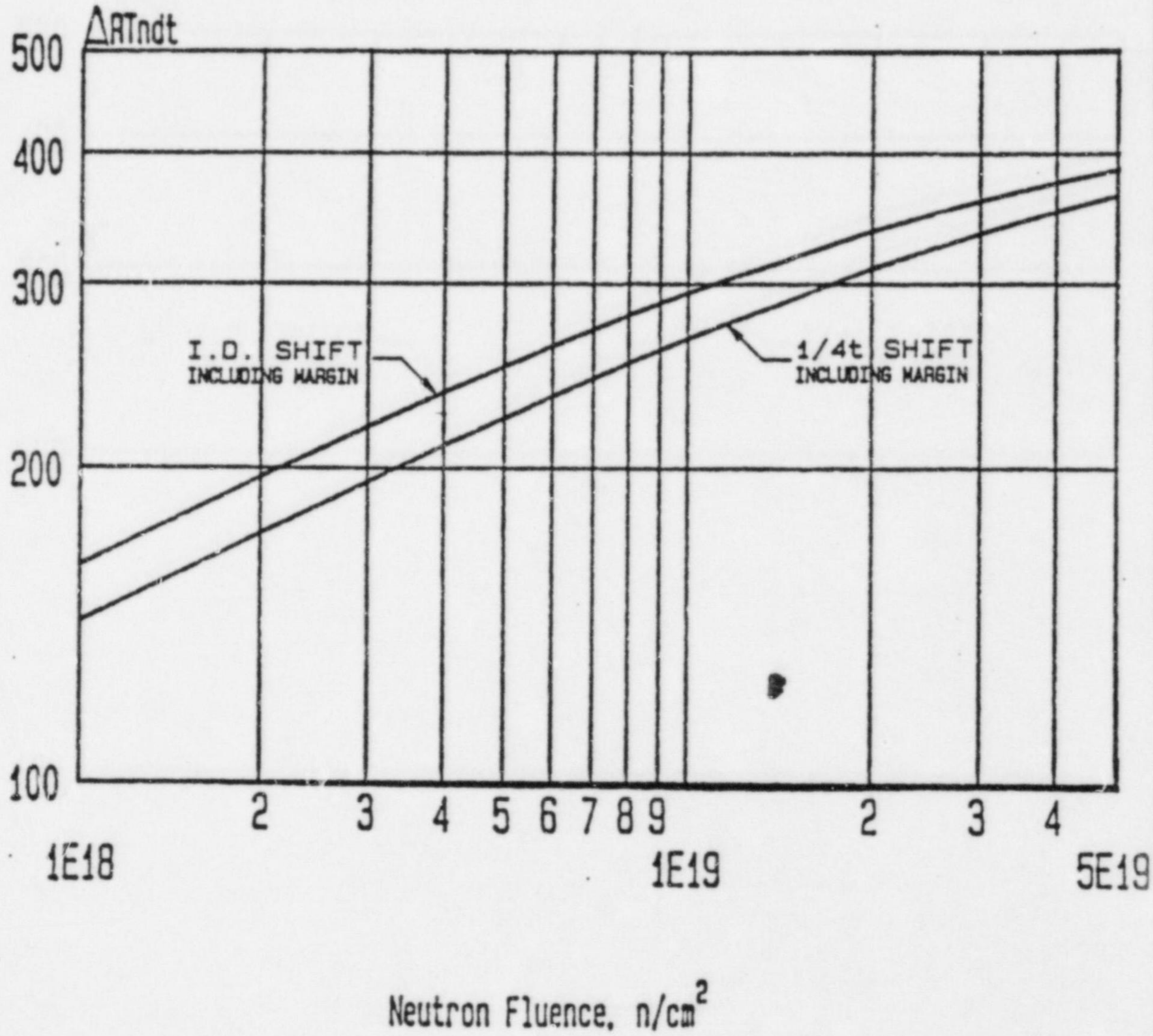

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FIGURE
2-3