LEBOEUF, LAMB, LEIBY & MACRAE

A PARTNERSHIP INCLUDING PROFESSIONAL CORPORATIONS

1333 NEW HAMPSHIRE AVENUE, N.W.
WASHINGTON, D.C. 20036

202-457-7500

TELEX 440274 TELECOPIER 202-457-7543

140 BROADWAY
NEW YORK, N.Y. 10005
212-269-1100
TELEX 423416

83 CENTRAL STREET
BOSTON, MA 02109
617-451-1385

336 FAYETTEVILLE STREET MALL
P.O. BOX 750
RALEIGH, NC 27602
919-833-9789

47 BERKELEY SQUARE
LONDON W1X 5DB, ENGLAND
TELEPHONE: 01-493-7331
TELEX: 25955

500 KEARNS BUILDING
136 SOUTH MAIN
SALT LAKE CITY, UT 84101
801-355-6918

411 PEQUOT AVENUE
SOUTHPORT, CT 06490
203-259-8383

March 7, 1983

Mr. Harold Denton
United States Nuclear
Regulatory Commission
7920 Norfolk Avenue
Phillips Building
Bethesda, MD 20014

Re: Omaha Public Power District
Fort Calhoun Station, Unit No. 1
Docket No. 50-285

Dear Mr. Denton:

As counsel for Omaha Public Power District, we hereby submit three (3) signed originals and nineteen (19) copies of a document entitled "Application for Amendment of Operating License", together with forty (40) copies of the proposed Technical Specifications. The application seeks to amend Sections 2.1.2 and figures 2-1A, 2-1B, 2-2A, and 2-2B of the Technical Specifications set forth in Appendix A to revise the reactor coolant system heatup and cooldown limits for operation through the end of fuel Cycle 8.

The proposed Amendment is deemed to be a Class III Amendment within the meaning of Section 170.22 of the regulations of the U.S. Nuclear Regulatory Commission. Accordingly, a check for the appropriate fee of $4,000 is also enclosed.

A Certificate of Service showing service of these documents on the persons listed therein is also enclosed.

A001
w/check
$4,000

We respectfully request that approval for this amendment be granted by no later than June 1, 1983.

Very truly yours,

Le Boeuf, Lamb, Leiby + MacRae

LeBoeuf, Lamb, Leiby & MacRae
Attorneys for Omaha Public
Power District

Enclosures

BEFORE THE UNITED STATES
NUCLEAR REGULATORY COMMISSION

In the matter of	)	
	)	Docket No. 50-285
OMAHA PUBLIC POWER DISTRICT	)	
(Fort Calhoun Station,	)	
Unit No. 1)	)	

CERTIFICATE OF SERVICE

I hereby certify that I have served a document entitled "Application for Amendment of Operating License", together with proposed changes to Technical Specifications by mailing a copy thereof first class, postage prepaid, to the following persons this 7th day of March, 1983.

Mr. Frank Gibson
Director
W. Dale Clark Library
215 South 15th Street
Omaha, NE 68102

Mr. Emmet Rogert
Chairman, Washington County
Board of Supervisors
16th & Colfax Streets
Blair, NE 68008

Marilyn Tebor Shaw

Marilyn Tebor Shaw
LeBoeuf, Lamb, Leiby & MacRae
Attorneys for Omaha Public Power District

(a) The curve in Figure 2-3 shall be used to predict the increase in transition tremperature 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 and the reduced vessel fluence rate provided by core load designs beginning with fuel Cycle 8, the predicted surface fluence at the reactor vessel belt-line weld material for 40 years at 1500 MWt and an 80% load factor is $3.1 \times 10^{19} n/cm^2$. 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 Figure 2-1A through 2-2B 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 162°F because components related to this temperature are also not subject to fast neutron flux.

(d) The minimum temperature at which the 100°F/hr cooldown rate curve may be used is defined by the LPSI pumps outlet pressure to provide for protection against low temperature overpressurization per Technical Specification 2.3(3). The Technical Specification 2.3(3) shall be revised each time the curves of Figures 2-1A through 2-2B 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 belt-line 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. In accordance with the methods identified in "NRC Staff Evaluation of Pressurized Thermal Shock", SECY 82-465, Appendix E, a weld material reference temperature (RT_{NDT}) was established at -22°F based on a mean value plus two standard deviations.

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 $3.1 \times 10^{19} n/cm^2$, including tolerance, 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 ($E \geq 1$ MeV) exposure of the reactor vessel including tolerance is computed to be $3.1 \times 10^{19} n/cm^2$ for 40 years

operation at 1500 MWt and 80% load factor.(5) The predicted T_{NDT} shift for an integrated fast neutron (E>1 MeV) exposure of $3.1 \times 10^{19} n/cm^2$ is 321°F, the value obtained from the curve shown in Figure 2-3. 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 first removed irradiated reactor vessel surveillance specimen, combined with a new core loading design for Cycle 8, indicates that the fluence at the end of 8.49 Effective Full Power Years (EFPY) at 1500 MWt will be $1.05 \times 10^{19} n/cm^2$ on the inside surface of the reactor vessel.(5) This results in a total shift of the RT_{NDT} of 260°F for the area of greatest sensitivity (weld metal) at the 1/4t location as determined from Figure 2-3. Operation through fuel Cycle 10 will result in less than 8.49 EFPY.

The limit lines in Figures 2-1A through 2-2B 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 belt-line. 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 $K_{IT} = 0$.

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

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.

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

M_T = 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 + 100°F + 12°F = 162°F. As indicated previously, an RT_{NDT} for all material with the exception of the reactor vessel belt-line was established at 50°F. ASME III, Art. NB-2332(b) 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 (.20)(3125) - 48 - 32 psi = 545 psia cannot be exceeded.

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

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. 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. Reactor Critical Heatup and Cooldown Figures. During low power physics testing, the reactor may be made critical at reduced temperature and pressure. To provide for heatup and cooldown during testing, Appendix G requires that the RCS temperature be increased an additional 40°F beyond heatup and cooldown curves for the non-critical reactor. Also, Appendix G requires that the RCS temperature must be greater than the minimum temperature, 348°F, required for the 2310 psia hydrostatic testing to 110% of the 2100 psia RCS operating pressure, in accordance with Article IWB-5000 of the ASME Boiler and Pressure Vessel Code, Section XI.

F. Minimum Temperature for 100°F/hr Cooldown Rate = 153°F. This limit provides protection against low temperature overpressurization during operation of the LPSI pumps.[6] This temperature corresponds to a pressure of 200 psia on the 100°F/hr curve, which is the LPSI pump dead head and minimum flow pressure. For temperatures of 153°F or less, a cooldown rate of 20°F/hr maximum will allow unrestricted operation of the LPSI pumps so that shutdown cooling may be utilized.

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

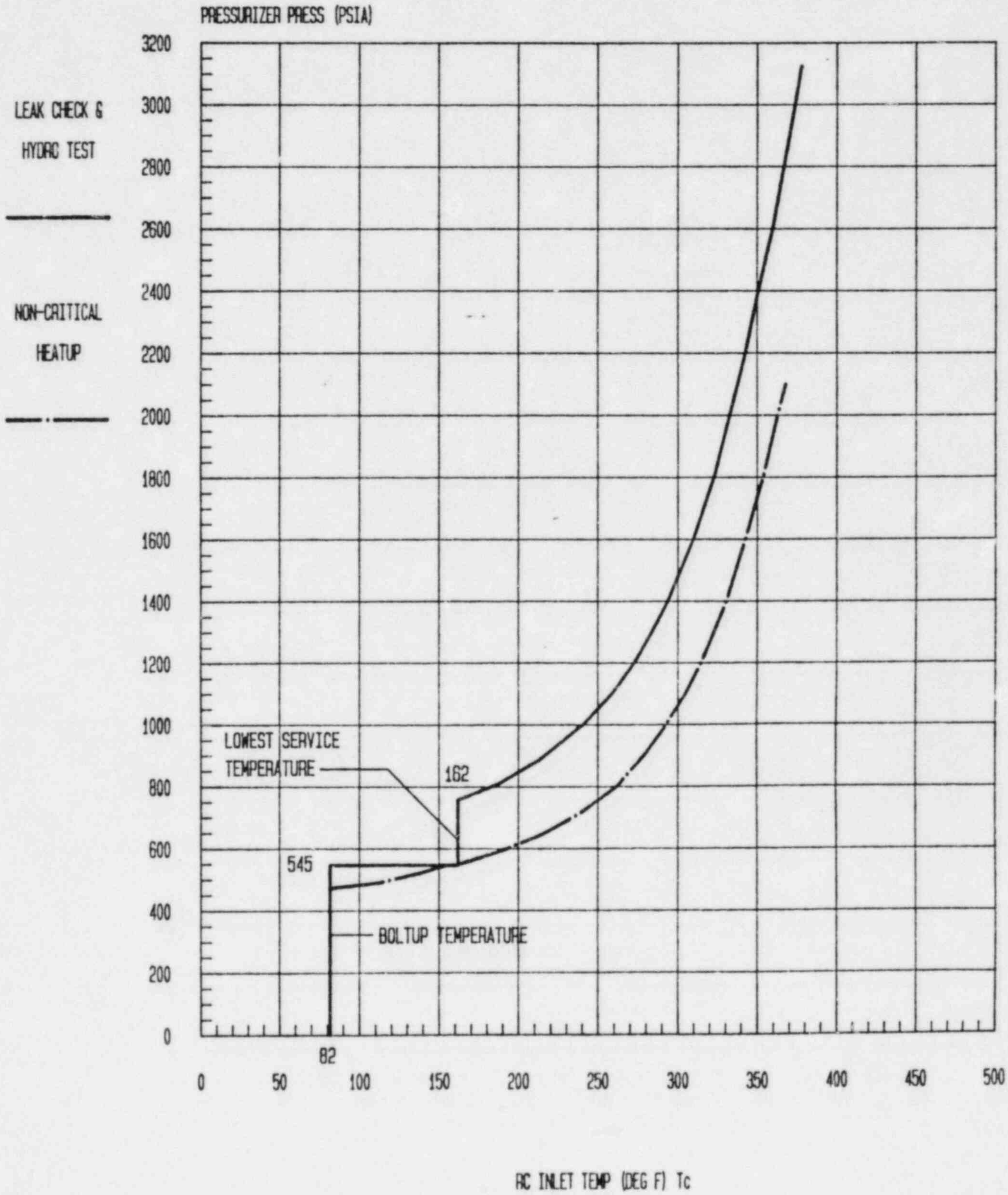
(6) Technical Specification 2.3(3)

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

RCS PRESS-TEMP LIMITS HEATUP 8.49 EFPY

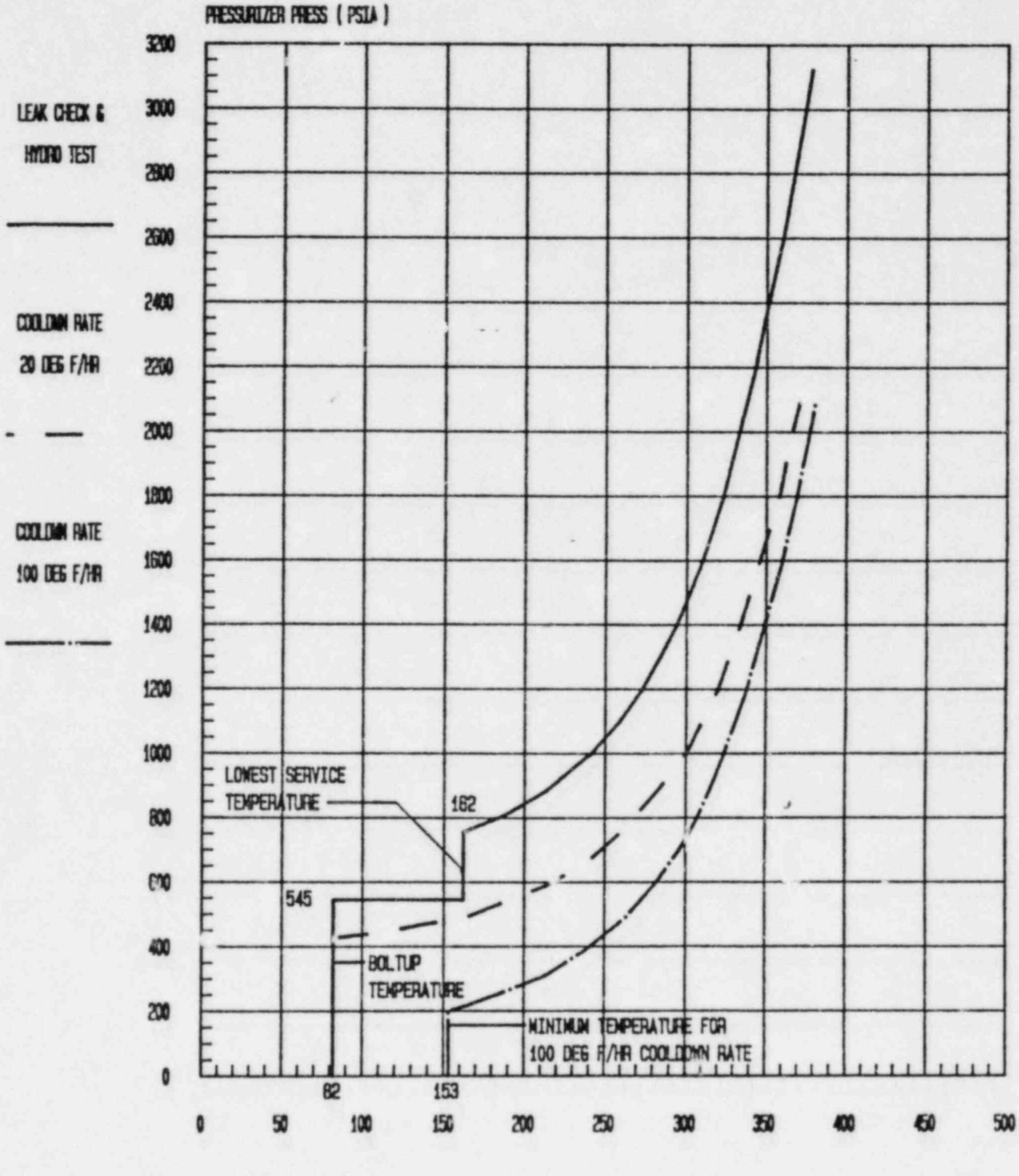
1500 MWt

REACTOR NOT CRITICAL



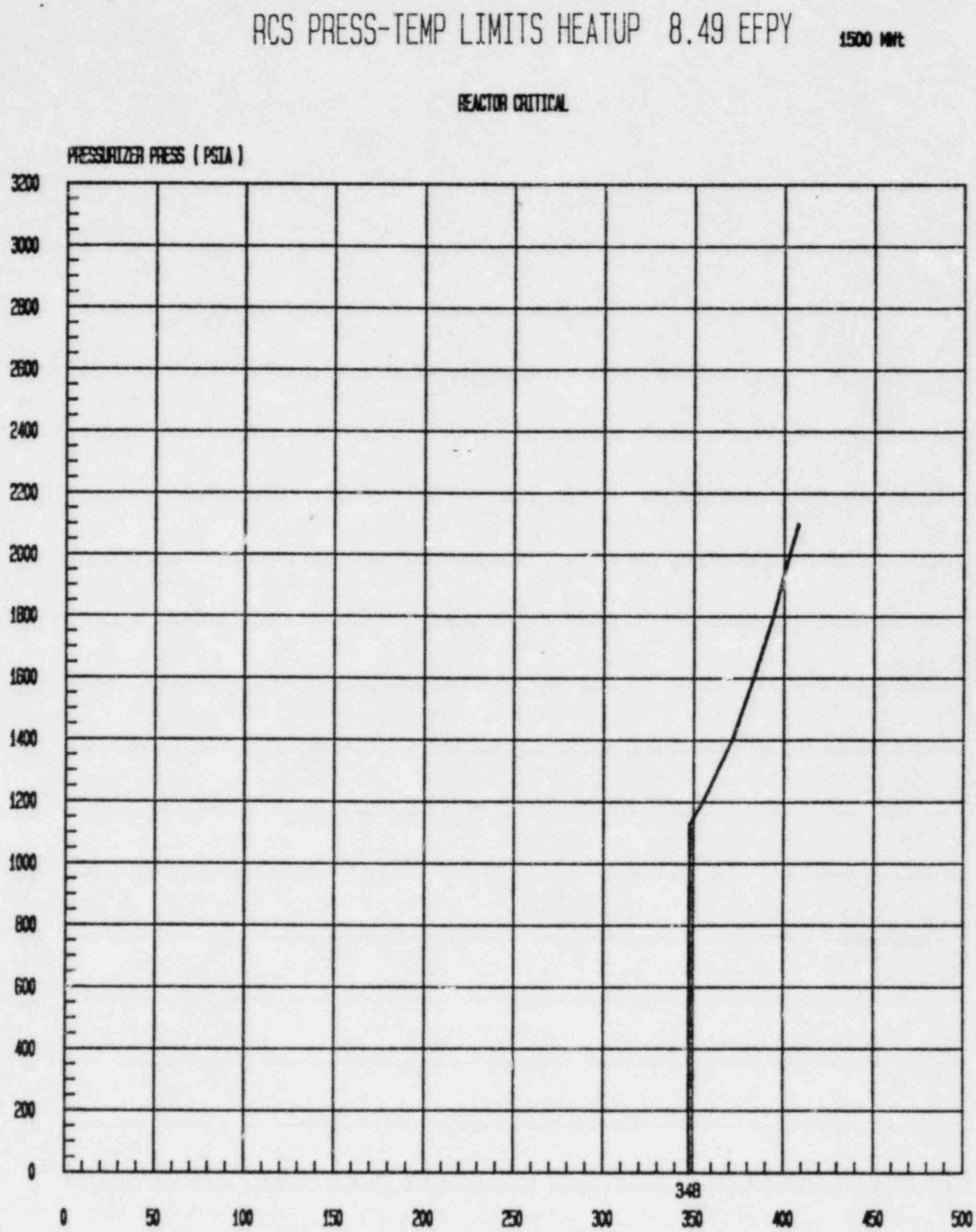
RCS PRESS-TEMP LIMITS COOLDOWN 8.49 EFPY 1500 MWt

REACTOR NOT CRITICAL



FORT CALHOUN
TECHNICAL
SPECIFICATIONS

FIGURE
2-1B

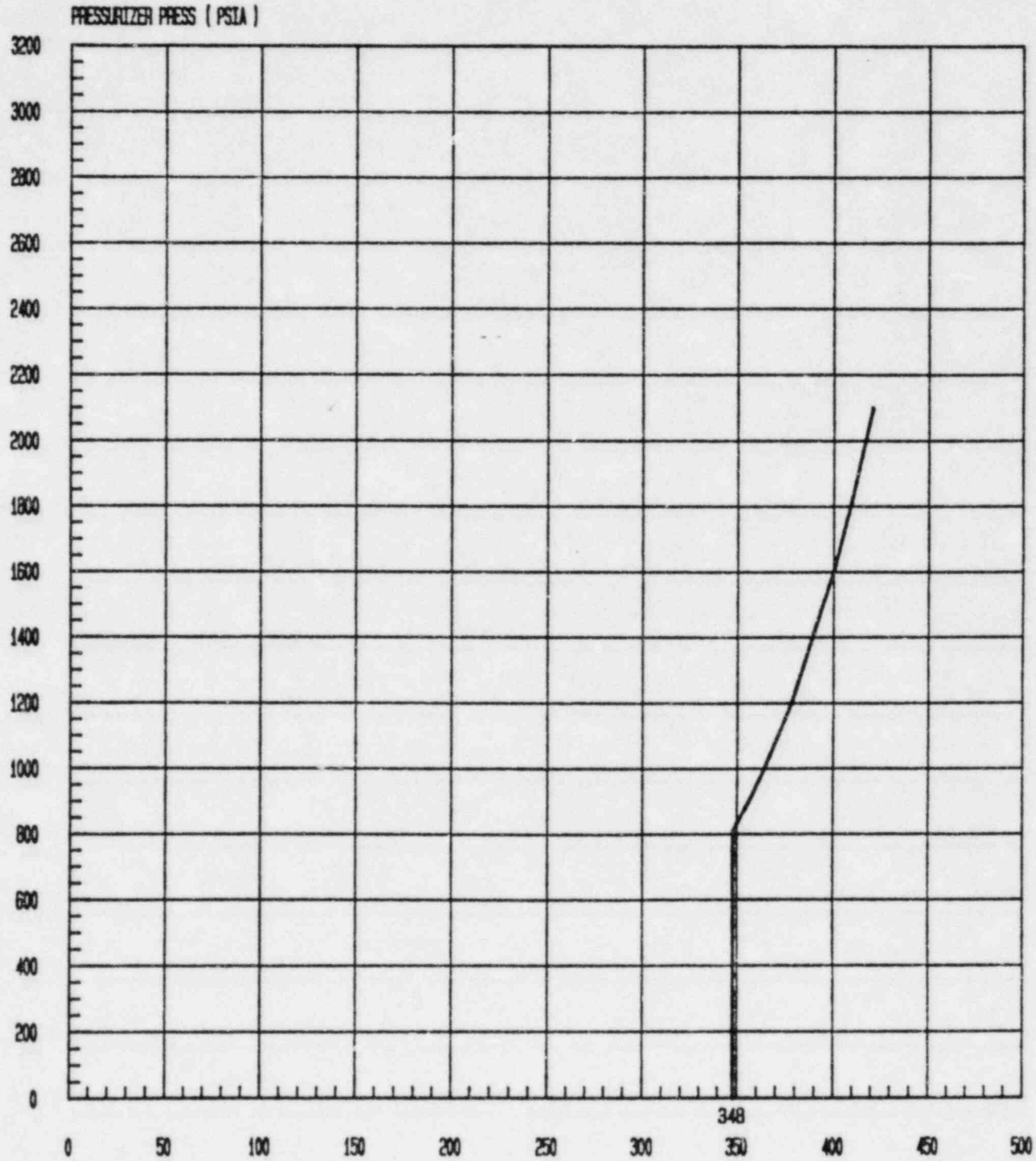


FORT CALHOUN
TECHNICAL
SPECIFICATIONS

FIGURE
2-2A

RCS PRESS-TEMP LIMITS COOLDOWN 8.49 EFPY 1500 MWt

REACTOR CRITICAL



FORT CALHOUN
TECHNICAL
SPECIFICATIONS

FIGURE
2-2B

DISCUSSION

This amendment application is required to allow for the safe operation of the Fort Calhoun Station reactor and associated primary coolant system beyond the 6.1 Effective Full Power Years (EFPY) of operation for which the present Technical Specifications (TS) are valid. This application requests continued operation through 8.49 EFPY.

In determining the limiting conditions for operation at this increased EFPY, the impact of the initial nil-ductility transition reference temperature (RT_{NDT}) and associated RT_{NDT} shift must be accounted for due to the effect of neutron fluence on the reactor vessel belt-line region welds. For previous cycles, the District conservatively utilized 0°F as the initial RT_{NDT}, as recommended by Branch Technical Position MTEB 5-2. However, recent evaluations performed by the Commission demonstrated that this initial RT_{NDT} value is overly conservative. Specifically, Appendix E of the report, "NRC Staff Evaluation of Pressurized Thermal Shock: SECY 82-465", states "Estimates (i.e. initial RT_{NDT} values) based on the 3 Charpy test results and MTEB 5-2 are not very satisfactory, because they are overconservative for some cases." The recommended NRC methodology in this report for computing RT_{NDT}, which utilizes a mean RT_{NDT} value obtained from generic weld data and ensures conservatism by adding a two sigma uncertainty value, results in an initial RT_{NDT} for a Combustion Engineering reactor vessel such as Fort Calhoun of -22°F. Therefore, this value will be utilized as the Fort Calhoun Station initial RT_{NDT} value in this and future cycle analyses.

Additionally, commencing with start-up from the present (1983) refueling outage, the District will utilize a new core loading pattern which will result in a significant decrease in fluence to the critical belt-line welds. Further discussion of this core loading pattern is provided in the District's letter dated January 27, 1983. The combination of the lower initial RT_{NDT} value and implementation of the reduced cycle fluence extends the present Technical Specification heatup and cooldown limits applicability through 8.49 EFPY, with minor changes as indicated in the revised specification. This amendment will provide pressure - temperature operating limits through Cycle 10.

Regulatory Guide 1.99, Revision 1, provided the methodology used to determine the nil-ductility transition reference temperature (RT_{NDT}) shift reflected in the proposed heatup and cooldown limit curves. The fluence value for the reactor vessel belt-line weld material was determined using the end-of-life predicted fluence of $4.4x10^{19}n/cm^2$. This value was calculated and approved for Cycle 6 operation using the Fort Calhoun Station first surveillance capsule test data as detailed in the Combustion Engineering report, "Evaluation of the Irradiated Capsule W-225", Revision 1, dated August 1980. In addition, the rate of fluence for Cycle 8 and future cycles was conservatively assumed to be 37% less than that of previous cycles due to the reduced fluence core loading pattern to be implemented for Cycle 8. Thus, the heatup and cooldown pressure-temperature limit curves used for Cycle 7 were updated to 8.49 EFPY without a temperature shift and yet still ensure adequate fracture toughness is maintained through all conditions of normal operation, including anticipated operation transients and system hydrostatic tests.

The current Technical Specification, limited to 6.1 EFPY, will provide operating limits for a period of 66 days of full power operation (2,367,000 MW-HRS) after initial criticality is achieved for fuel Cycle 8. Therefore, Commission approval of the proposed Technical Specifications by no later than May 30, 1983 is requested.

JUSTIFICATION FOR FEE CLASSIFICATION

The proposed amendment is deemed to be Class III, within the meaning of 10 CFR 170.22, in that it involves a single safety concern.

ATTACHMENT C