2.0 LIMITING CONDITIONS FOR OPERATION
2.1 Reactor Coolant System (Continued)

2.1.2 Heatup and Cooldown Rate

Applicability

Applies to the temperature change rates and pressure of the reactor coolant system.

Objective

To specify limiting conditions of the reactor coolant system heatup and cooldown rates.

Specification

The reactor coolant pressure shall be limited during plant operation in accordance with Figure 2-1A and 2-1B and as follows:

(1) The heatup rate shall not exceed 100°F in any one hour period.

(2) Allowable combinations of pressure and temperature (Tc) for a specific cooldown rate shall be below and to the right of the applicable limit lines as shown on Figures 2-1A and 2-1B.

(3) The heatup rate of the pressurizer shall not exceed 100°F in any one hour period.

(4) The cooldown rate of the pressurizer shall not exceed 200°F in any one hour period.

(5) When any of the above limits are exceeded, the following corrective actions shall be taken:

(a) Immediately initiate action to restore the temperature or pressure to within the limit.

(b) Perform an analysis to determine the effects of the out of limit condition on the fracture toughness properties of the reactor coolant system.

(c) Determine that the reactor coolant system remains acceptable for continued operation or be in cold shutdown within 36 hours.

(6) Before the radiation exposure of the reactor vessel exceeds the exposure for which they apply, Figures 2-1A and 2-1B shall be updated in accordance with the following criteria and procedures:

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 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 3.1×10^{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 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 162°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 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≥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 3.1×10^{19} n/cm^2, including tolerance at the vessel inside surface, 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 including tolerance is computed to be 3.1×10^{19} n/cm^2 at the vessel inside surface for 40 years operation at 1500 MWt and 80% load factor. The exposure at the 1/4t depth from the inner surface is computed to be 1.9×10^{19} n/cm^2.[5] The predicted T_{NDT} shift for an integrated fast neutron (E≥1 MeV) exposure of 1.9×10^{19} n/cm^2 is 321°F, the value obtained from the curve shown

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

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 7.0 Effective Full Power Years (EFPY) at 1500 MWt will be 9.15×10^{18} n/cm^2 on the inside surface of the reactor vessel and 5.65×10^{18} n/cm^2 at the 1/4t depth.(5) This results in a total shift of the RT_{NDT} of 253°F for the area of greatest sensitivity (weld metal) at the 1/4t location as determined from Figure 2-3. Operation through fuel Cycle 8 will result in less than 7.0 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 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 $K_{IT} = 0$.

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.

References

(1) USAR, Section 4.2.2

(2) ASME Boiler and Pressure Vessel Code, Section III

(3) USAR, Section 4.2.4

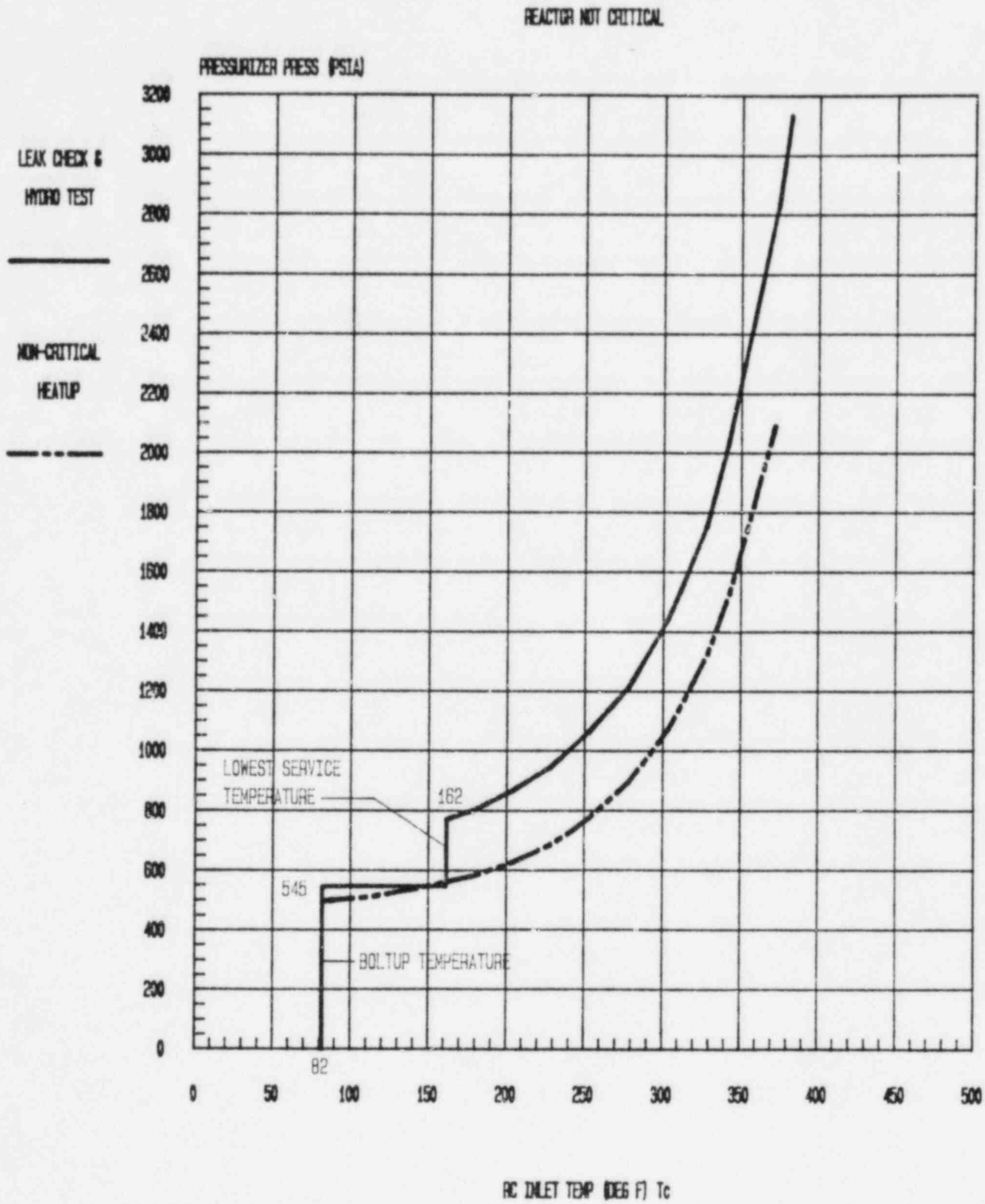
(4) USAR, Section 3.4.6

(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 7.0 EFPY 1500MWt

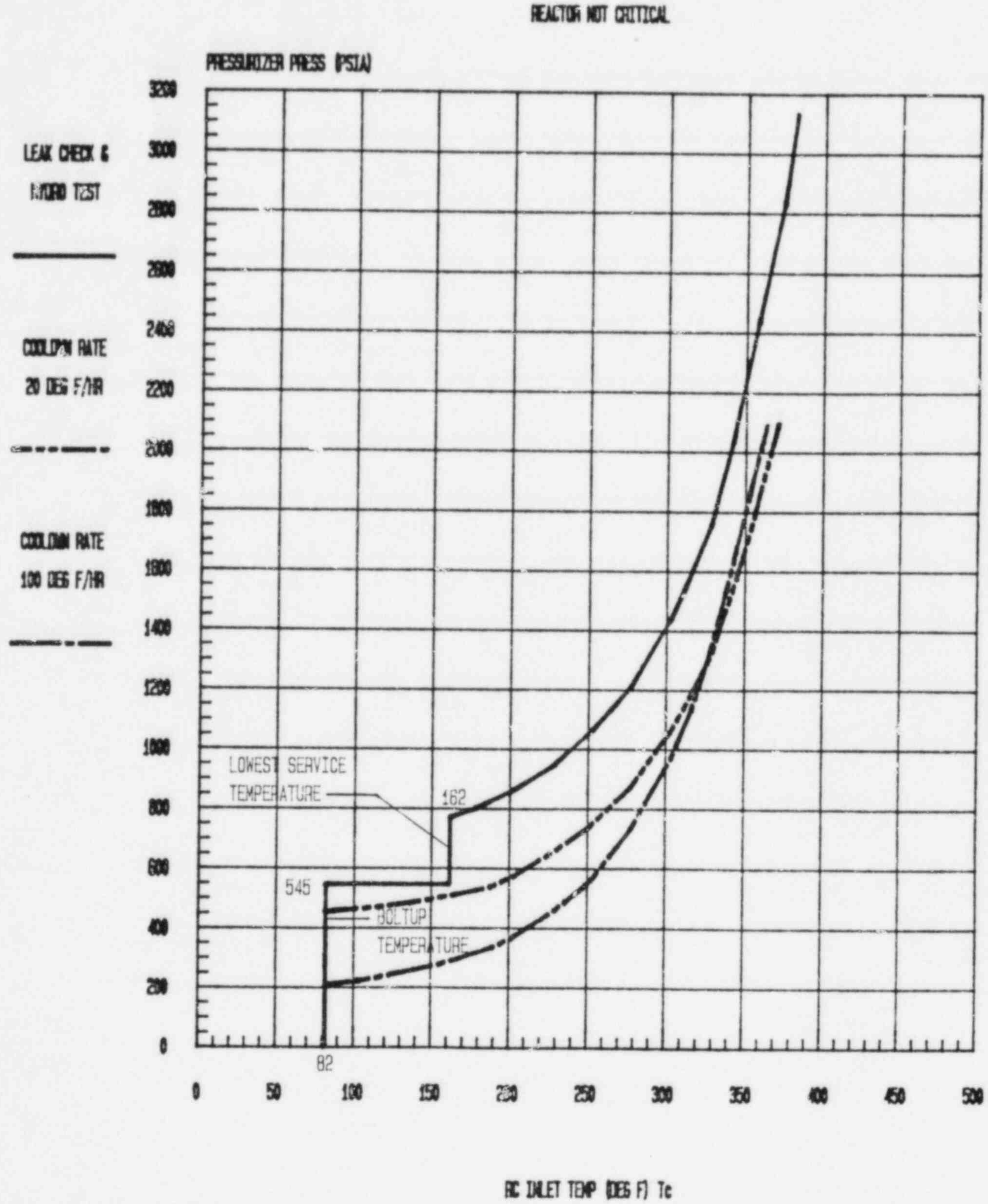


FORT CALHOUN
TECHNICAL
SPECIFICATIONS

FIGURE
2-1A

RCS PRESS-TEMP LIMITS COOLDOWN 7.0 EFPY

1500MWt



FORT CALHOUN
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FIGURE
2-1B

FIGURE 2-2A

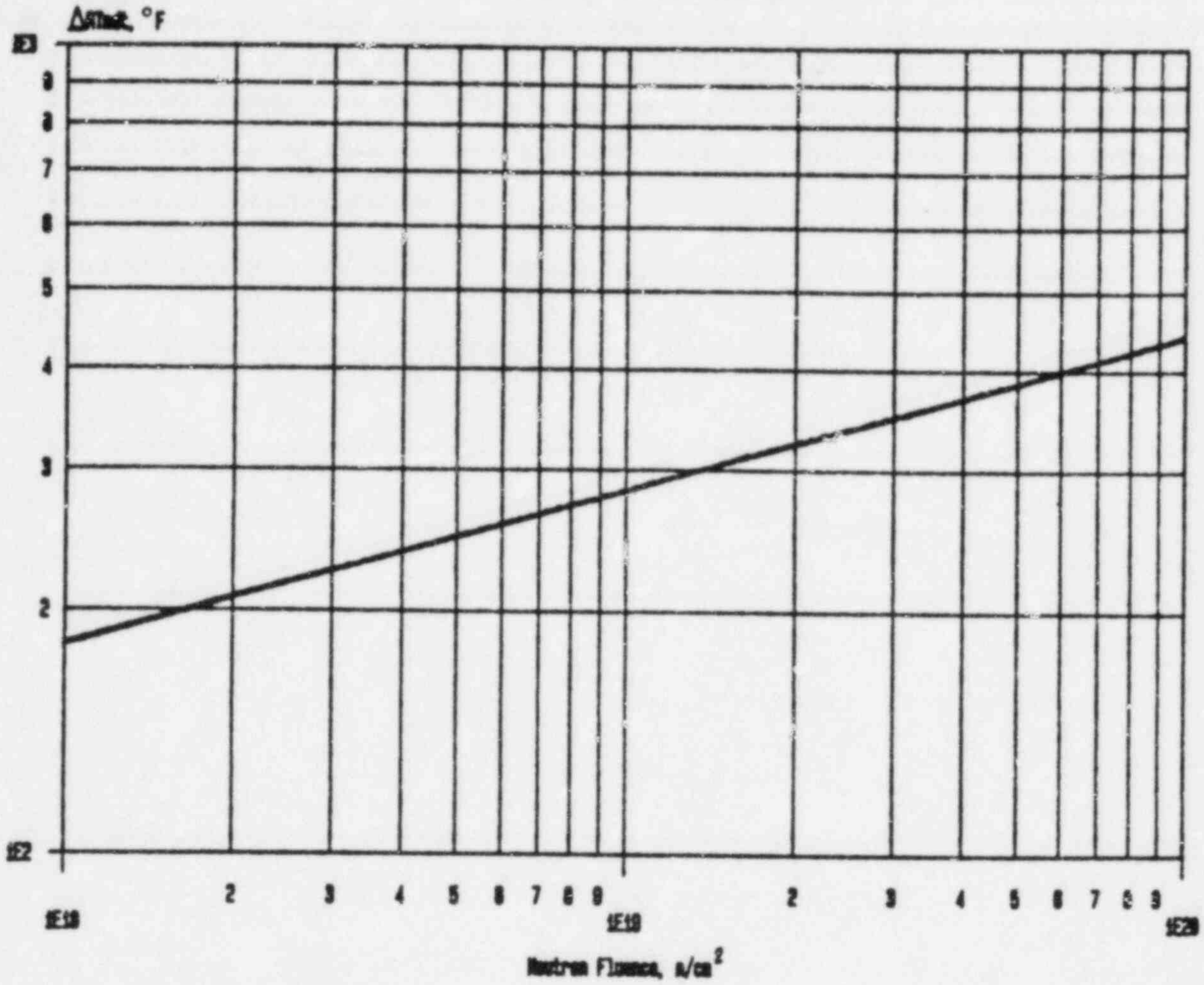
DELETED

FIGURE 2-2B

DELETED

PREDICTED RADIATION INDUCED NDTT SHIFT

Fort Calhoun Reactor Vessel Beltline



(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 309°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 shut down cooling, the valving will be changed and must be properly aligned prior to start-up of the reactor.

2.0 LIMITING CONDITIONS FOR OPERATION

2.3 Emergency Core Cooling System (Continued)

The operable status of the various systems and components is to be demonstrated by periodic tests. A large fraction of these tests will be performed while the reactor is operating in the power range.

If a component is found to be inoperable, it will be possible in most cases to effect repairs and restore the system to full operability within a relatively short time. For a single component to be inoperable does not negate the ability of the system to perform its function. If it develops that the inoperable component is not repaired within the specified allowable time period, or a second component in the same or related system is found to be inoperable, the reactor will initially be put in the hot shutdown condition to provide for reduction of cooling requirements after a postulated loss-of-coolant accident. This will also permit improved access for repairs in some cases. After a limited time in hot shutdown, if the malfunction(s) is not corrected, the reactor will be placed in the cold shutdown condition utilizing normal shutdown and cooldown procedures. In the cold shutdown condition, release of fission products or damage of the fuel elements is not considered possible.

The plant operating procedures will require immediate action to effect repairs of an inoperable component and therefore in most cases repairs will be completed in less than the specified allowable repair times. The limiting times to repair are intended to assure that operability of the component will be restored promptly and yet allow sufficient time to effect repairs using safe and proper procedures.

The requirement for core cooling in case of postulated loss-of-coolant accident while in the hot shutdown condition is significantly reduced below the requirements for a postulated loss-of-coolant accident during power operation. Putting the reactor in the hot shutdown condition reduces the consequences of a loss-of-coolant accident and also allows more free access to some of the engineered safeguards components in order to effect repairs.

Failure to complete repairs within 48 hours of going to the hot shutdown condition is considered indicative of a requirement for major maintenance and, therefore, in such a case, the reactor is to be put into the cold shutdown condition.

With respect to the core cooling function, there is functional redundancy over most of the range of break sizes.(3)(4)

The LOCA analysis confirms adequate core cooling for the break spectrum up to and including the 32 inch double-ended break assuming the safety injection capability which most adversely affects accident consequences and are defined as follows. The entire contents of all four safety injection tanks are assumed to 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 USAR Figures 1-9 (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 309°F on Figure 2-1B. Thus, at least two HPSI pumps will be disabled at 309°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.

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

References

(1) USAR, Section 14.15.1

(2) USAR, Section 6.2.3.1

(3) USAR, Section 14.15.3

(4) USAR, Appendix K

(5) Omaha Public Power District's Submittal, December 1, 1976

(6) Technical Specification 2.1.2, Figure 2-1B

DISCUSSION

This amendment application is required to allow for safe operation of the Fort Calhoun Station reactor and associated primary coolant system beyond 6.1 Effective Full Power Years (EFPY) of operation for which the current Technical Specifications (TS) are valid. This application requests continued operation through 7.0 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 welds in the reactor vessel belt-line region. 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 method, and thus the 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 two sigma uncertainty values, results in an initial RT_{NDT} for a Combustion Engineering reactor vessel such as Fort Calhoun of -22°F. As a result of this information, the District requested and received new non-irradiated baseline curves for weld material with a -22°F initial RT_{NDT} from Combustion Engineering for use in this and future cycle analyses.

As noted in the Technical Specifications, when addressing plant cooldown the thermal and pressure stresses are additive in determining the reference stress intensity factor K_{IR} and the corresponding temperature at the 1/4t location (Figure G-2110-1 of ASME III). This corresponding temperature is subsequently corrected to the reactor vessel fluid temperature in establishing limiting temperatures, thereby resulting in a reduction of about 20°F for the 100° per hour cooldown case. The reduction is based on a calculated thermal gradient through the wall in which the RV wall temperature lags the RV fluid temperature during a cooldown. A lesser reduction can be seen for the 20° per hour cooldown curve because the thermal gradient through the wall is smaller for the lower cooldown rate.

For the case of heatup, the thermal stresses are opposite in sign from the pressure stresses. Therefore, it is conservative, when calculating K_{IR}, to assume an isothermal heatup (K_{IT} equal zero). A correction from the 1/4t location to the reactor vessel fluid is not made for the heatup case because there is no thermal gradient through the RV wall for the isothermal condition. If thermal stresses were taken into consideration for heatup (non-isothermal), the RV fluid temperature would be higher than the RV 1/4t wall temperature because the vessel wall lags the fluid temperature during a change in fluid system temperature. The baseline curves, however, are based on isothermal heatup and are found to be conservative as compared to similar heatup curves that account for thermal stresses and a fluid temperature correction due to the temperature gradient in the RV wall.

Additionally, commencing with start-up from the present (1983) refueling outage, the District will utilize a new low leakage core loading pattern which will result in a significant decrease in fluence to the critical belt-line welds. Further discussion of this core loading patter is provided in the District's letter dated January 27, 1983.

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 4.4×10^{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 newly developed baseline curves received from Combustion Engineering were shifted to produce heatup and cooldown pressure-temperature limit curves which reflect the predicted fluence expected through 7.0 EFPY and ensure adequate fracture toughness is maintained through all conditions of normal operation, including anticipated operation transients and system hydrostatic tests.

The current TS, limited to 6.1 EFPY, will provide operating limits for a period of 66 days of full power operation (2,376,000 MW-HRs) after initial criticality is achieved for fuel Cycle 8. Therefore, Commission approval of the proposed TS is requested by no later than May 30, 1983.

Figures 2-2A and 2-2B for critical heatup and cooldown are deleted. TS 2.10.1(1) restricts critical power operation to above 515^{o}F; thus, Figures 2-2A and 2-2B are no longer required.