TABLE OF CONTENTS (Continued)

Amendment No. ~~32,34,43,54,55,57,~~
~~73,80,86,89,93,99,141,152,157,184,~~ 185

TECHNICAL SPECIFICATIONS - FIGURES

TABLE OF CONTENTS

DEFINITIONS

Dose Equivalent I-131 (μCi/gm) = μCi/gm of I-131
+ 0.0361 x μCi/gm of I-132
+ 0.270 x μCi/gm of I-133
+ 0.0169 x μCi/gm of I-134
+ 0.0838 x μCi/gm of I-135

$\bar{E}$ - Average Disintegration Energy

$\bar{E}$ is the average (weighted in proportion to the concentration of each radionuclide in the reactor coolant at the time of sampling) of the sum of the average beta and gamma energies per disintegration, in MEV, for isotopes, other than iodines, with half lives greater than 15 minutes making up at least 95% of the total non-iodine radioactivity in the coolant.

Offsite Dose Calculation Manual (ODCM)

The document(s) that contain the methodology and parameters used in the calculations of offsite doses resulting from radioactive gaseous and liquid effluents, in the calculation of gaseous and liquid effluent radiation monitoring Warn/High (trip) Alarm setpoints, and in the conduct of the Environmental Radiological Monitoring Program. The ODCM shall also contain:

1) The Radiological Effluent Controls and the Radiological Environmental Monitoring Program required by Specification 5.16.

2) Descriptions of the information that should be included in the Annual Radiological Environmental Operating Reports and Annual Radioactive Effluent Release Reports required by Specifications 5.9.4.a and 5.9.4.b.

Unrestricted Area

Any area at or beyond the site boundary access to which is not controlled by the licensee for purposes of protection of individuals from exposure to radiation and radioactive materials.

Core Operating Limits Report (COLR)

The Core Operating Limits Report (COLR) is a Fort Calhoun Station Unit No. 1 specific document that provides core operating limits for the current operating cycle. These cycle-specific core operating limits shall be determined for each reload cycle in accordance with Section 5.9.5. Plant operation within these operating limits is addressed in the individual specifications.

RCS Pressure-Temperature Limits Report (PTLR)

The RCS PRESSURE-TEMPERATURE LIMITS REPORT (PTLR) is a fluence dependent report providing Limiting Conditions for Operation for heatup, cooldown, inservice hydrostatic and leak testing, and core criticality limits in the form of Pressure-Temperature (P-T) limits to ensure prevention of brittle fracture. In addition, this report establishes Limiting Conditions for Operation which provide Low Temperature Overpressure Protection (LTOP) to assure the P-T limits are not exceeded during the most limiting LTOP event. The P-T limits and LTOP criteria in the PTLR are applicable through the Effective Full Power Years (EFPY) specified in the PTLR. NRC and ASME approved methodologies are used as the basis for the LCOs provided in the PTLR.

References

(1) USAR, Section 7.2

(2) USAR, Section 7.3

2.0 **LIMITING CONDITIONS FOR OPERATION**
2.1 Reactor Coolant System (Continued)
2.1.1 Operable Components (Continued)

(5) DELETED

(6) Both steam generators shall be filled above the low steam generator water level trip set point and available to remove decay heat whenever the average temperature of the reactor coolant is above 300°F. Each steam generator shall be demonstrated operable by performance of the inservice inspection program specified in Section 3.17 prior to exceeding a reactor coolant temperature of 300°F.

(7) Maximum reactor coolant system hydrostatic test pressure shall be 3125 psia. A maximum of 10 cycles of 3125 psia hydrostatic tests are allowed.

(8) Reactor coolant system leak and hydrostatic test shall be conducted within the limitations of the PTLR ~~Figures 2-1A and 2-1B~~.

(9) Maximum secondary hydrostatic test pressure shall not exceed 1250 psia. A minimum measured temperature of 73°F is required. Only 10 cycles are permitted.

(10) Maximum steam generator steam side leak test pressure shall not exceed 1000 psia. A minimum measured temperature of 73°F is required.

(11) If no reactor coolant pumps are operating, a non-operating reactor coolant pump shall not be started while T_c is below ~~385°F~~ the temperature listed in the PTLR unless at least one of the following conditions is met:

2.0 **LIMITING CONDITIONS FOR OPERATION**

2.1 Reactor Coolant System (Continued)

2.1.1 Operable Components (Continued)

(a) A minimum pressurizer steam space as specified in the PTLR ~~of 53% by volume or greater (50.6% or less actual level)~~ exists, or

(b) The temperature difference between the steam generator secondary side and the reactor coolant system cold leg ~~temperature~~ is less than the magnitude specified in the PTLR ~~30°F above that of the reactor coolant system cold leg~~.

(12) Reactor Coolant System Pressure Isolation Valves

(a) The integrity of all pressure isolation valves listed in Table 2.9 shall be demonstrated, except as specified in (b). Valve leakage shall not exceed the amounts indicated.

(b) In the event that the integrity of any pressure isolation valve specified in Table 2-9 cannot be demonstrated, reactor operation may continue, provided that at least two valves in each high pressure line having a nonfunctional valve are in and remain in the mode corresponding to the isolated condition. Manual valves shall be locked in the closed position; motor operated valves shall be placed in the closed position and power supply deenergized.

(c) If Specifications (a) and (b) above cannot be met, an orderly shutdown shall be initiated and the reactor shall be in the cold shutdown condition within 24 hours.

Basis

The plant is designed to operate with both reactor coolant loops and associated reactor coolant pumps in operation and maintain DNBR above 1.18 during all normal operations and anticipated transients.

When Specification 2.1.1(2) is applicable, the reactor coolant pumps (RCPs) are used to provide forced circulation heat removal during heatup and cooldown. Under these conditions, decay heat removal requirements are low enough that a single reactor coolant system (RCS) loop with one RCP is sufficient to remove core decay heat. However, two RCS loops are required to be OPERABLE to provide redundant paths for decay heat removal. Only one RCP needs to be OPERABLE to declare the associated RCS loop operable. Reactor coolant natural circulation is not normally used but is sufficient for core cooling. However, natural circulation does not provide turbulent flow conditions. Therefore, boron reduction in natural circulation is prohibited because mixing to obtain a homogeneous concentration in all portions of the RCS cannot be assured.

2.0 **LIMITING CONDITIONS FOR OPERATION**

2.1 Reactor Coolant System (Continued)

2.1.1 Operable Components (Continued)

When Specification 2.1.1(3) is applicable, a single reactor coolant loop or shutdown cooling loop provides sufficient heat removal capability for removing decay heat, but single failure considerations require that at least two loops be operable. Thus, if the reactor coolant loops are not OPERABLE, this specification requires two shutdown cooling pumps to be OPERABLE.

One of the conditions for which Specification 2.1.1(3) is applicable is when the RCS temperature (T_{cold}) is less than 210°F, fuel is in the reactor, and all reactor vessel closure bolts are fully tensioned. As soon as a reactor vessel head closure bolt is loosened, Specification 2.1.1(3) no longer applies, and Specification 2.8 is applicable. Specification 2.8 also requires two shutdown cooling loops to be operable if there is less than 23 feet of water above the top of the core.

The restrictions on availability of the containment spray pumps for shutdown cooling service ensure that the SI/CS pumps' suction header piping is not subjected to an unanalyzed condition in this mode. Analysis has determined that the minimum required RCS vent area is 47 in². This requirement may be met by removal of the pressurizer manway which has a cross-sectional area greater than 47 in².

When reactor coolant boron concentration is being changed, the process must be uniform throughout the reactor coolant system volume to prevent stratification of reactor coolant at lower boron concentration which could result in a reactivity insertion. Sufficient mixing of the reactor coolant is assured if one low pressure safety injection pump or one reactor coolant pump is in operation. The low pressure safety injection pump will circulate the reactor coolant system volume in less than 35 minutes when operated at rated capacity. The pressurizer volume is relatively inactive; therefore, it will tend to have a boron concentration higher than the rest of the reactor coolant system during a dilution operation. Administrative procedures will provide for use of pressurizer sprays to maintain a nominal spread between the boron concentration in the pressurizer and the reactor coolant system during the addition of boron.[1]

Both steam generators are required to be filled above the low steam generator water level trip set point whenever the temperature of the reactor coolant is greater than the design temperature ~~of the shutdown cooling system~~ to assure ~~a~~ redundant heat removal system for the reactor.

The LTOP enable temperature is documented in the PTLR ~~has been established at T_c = 385°F~~. The pressure transient analyses demonstrate that a single PORV is capable of mitigating overpressure events. Additional uncertainties have been applied to the Pressure-Temperature (P-T) limits to account for the case where a PORV is not available. ~~(T_c>385°F) which is the reason for the discontinuity in the P-T Figures.~~ ~~The curves have been conservatively smoothed for operations use.~~

2.0 **LIMITING CONDITIONS FOR OPERATION**
2.1 Reactor Coolant System (Continued)
2.1.1 Operable Components (Continued)

The design cyclic transients for the reactor system are given in USAR Section 4.2.2. In addition, the steam generators are designed for additional conditions listed in USAR Section 4.3.4. Flooded and pressurized conditions on the steam side assure minimum tube sheet temperature differential during leak testing. The minimum temperature for pressurizing the steam generator steam side is 70°F; in measuring this temperature, the instrument accuracy must be added to the 70°F; limit to determine the actual measured limit. The measured temperature limit will be 73°F based upon use of an instrument with a maximum inaccuracy of ± 2°F and an additional 1°F safety margin.

Formation of a ~~53%~~ steam space of the magnitude specified in the PTLR ensures that the resulting pressure increase would not result in any overpressurization should the first reactor coolant pump be started when the steam generator secondary side temperature is greater than that of the RCS cold leg. The steam space requirement is not applicable to the start of a reactor coolant pump if one or more pumps are in operation.

For the case in which the pressurizer steam space is less than the magnitude specified in the PTLR ~~53%~~, limitation of the steam generator secondary side/RCS cold leg ΔT to less than the magnitude specified in the PTLR ~~30°F~~ ensures that a single low setpoint PORV would prevent an overpressurization due to actuation of the first reactor coolant pump. This requirement is not applicable to the start of a reactor coolant pump if one or more pumps are operating.

References

(1) USAR Section 4.3.7

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 (RCS).

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:~~ The combination of RCS pressure, RCS temperature and RCS heatup and cooldown rates shall be maintained within the limits specified in the PTLR and as designated below:

~~(1)~~a. Allowable combinations of pressure and temperature (T_c) for a specific heatup rate shall be below and to the right of the applicable limit lines as shown in the PTLR ~~on Figure 2-1A~~.

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

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

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

Required Actions

(~~5~~1) 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:~~

~~(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). 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 allowable heatup/cooldown rates and cyclic operation. Cycle dependent information such as the pressure-temperature (P-T) limit curves and low temperature overpressure protection (LTOP) system limits are contained in the Fort Calhoun Station RCS Pressure - Temperature Limits Report (PTLR), which was developed using the methodologies of CE NPSD-683.[10]

~~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 Nonductile 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 nilductility 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 ($E \geq 1$ 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~~

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 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 fluence core loading designs initiated in Cycle 8, indicated that the fluence at the end of 20.0 Effective Full Power Years (EFPY) at 1500 MWt will be 1.50×10^{19} n/cm^2 on the inside surface of the reactor vessel. This results in a total shift of the RT_{NDT} of 298°F, including margin, for the area of greatest sensitivity (weld metal) at the 1/4t location as determined from Figure 2-3 and a shift of 241°F at the 3/4t location. Operation through fuel Cycle 19 will result in less than 20.0 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.~~

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

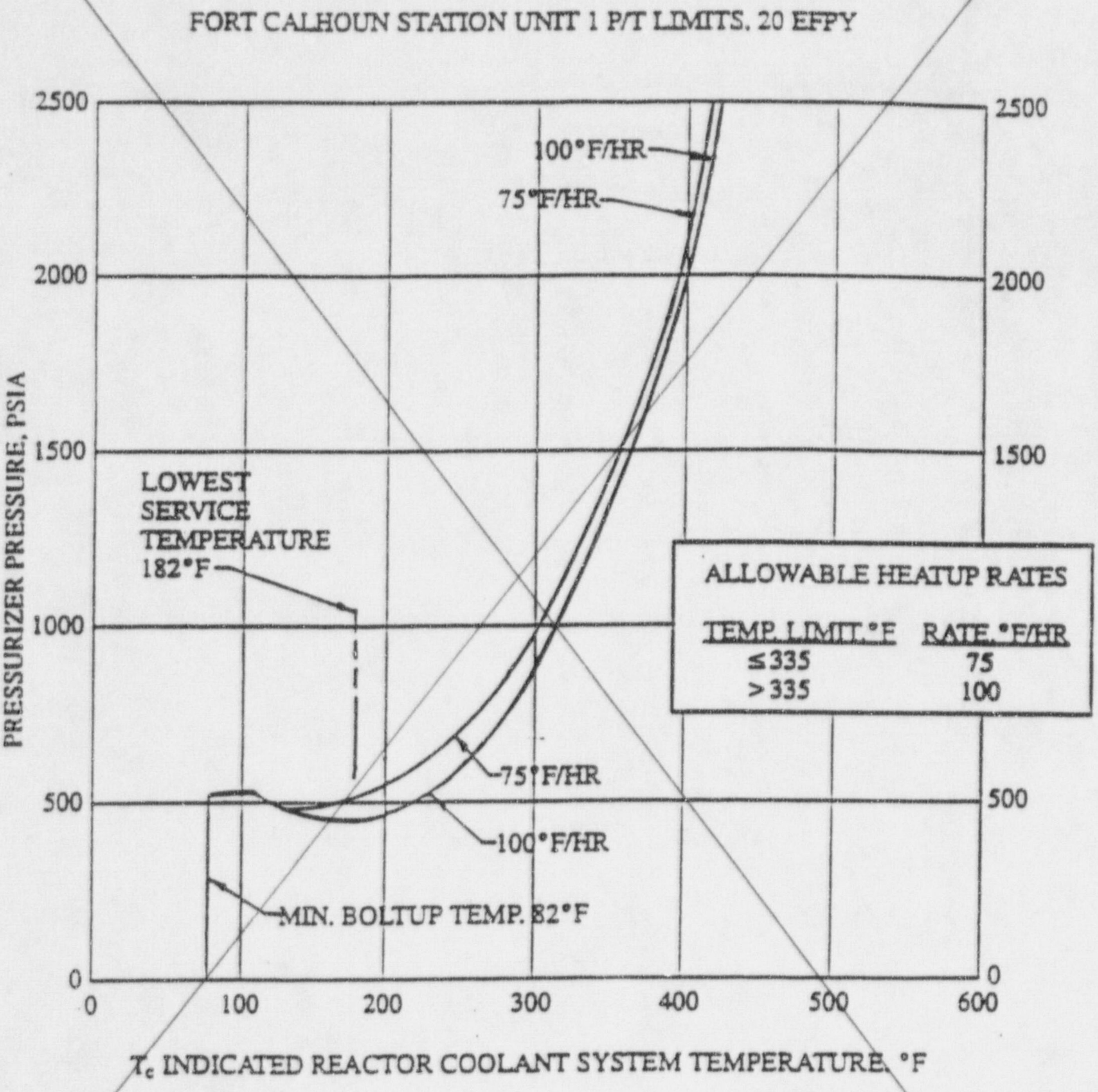
~~K_{IR} = Allowance stress intensity factor at temperature related to RT_{NDT} (ASME III Figure G-2110.1).~~

~~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.~~

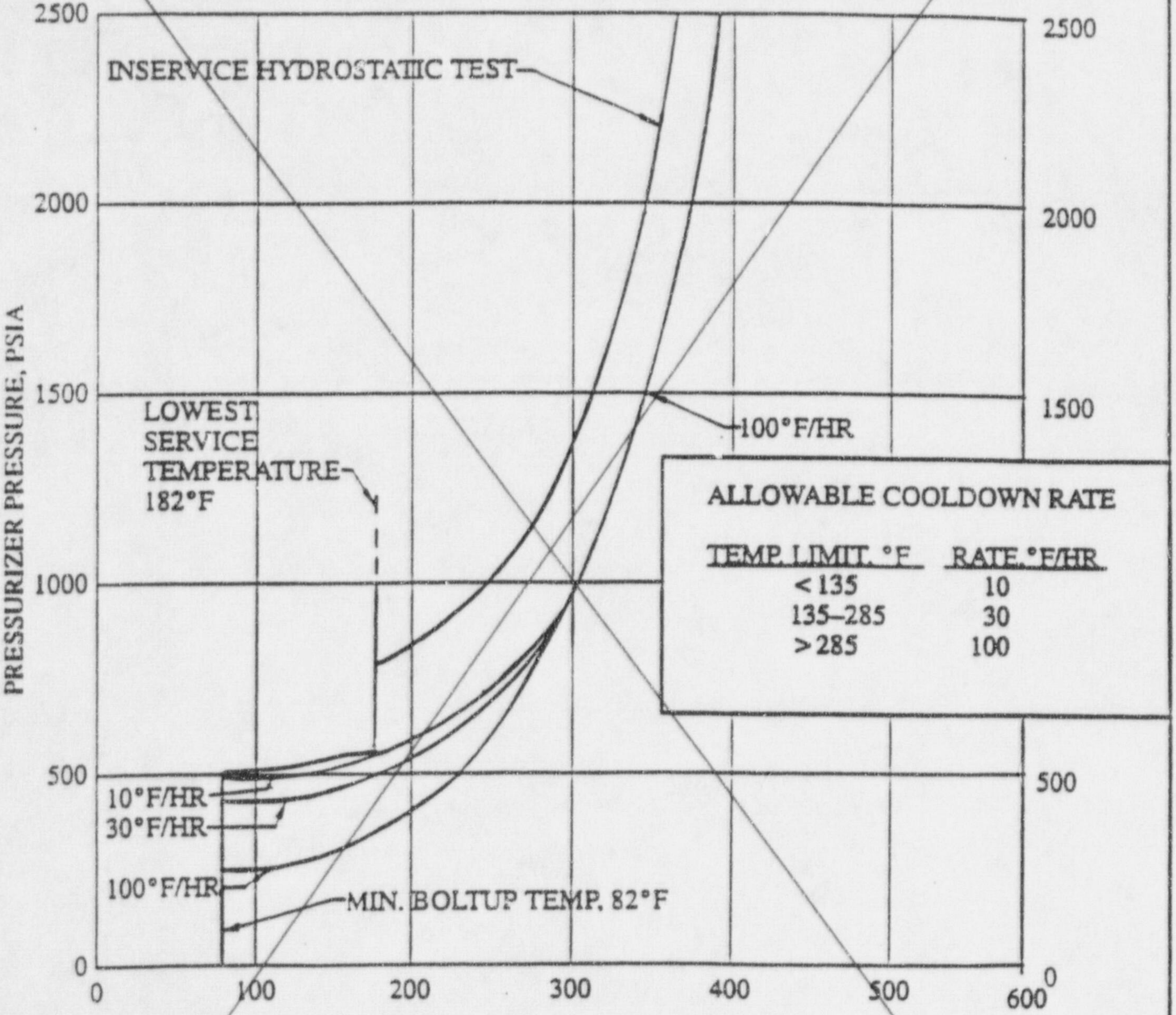
~~The above equation is applied to the reactor vessel beltline. For plant heatup the reference stress intensity in calculated for both the 1/4t and 3/4t locations. Composite curves are then generated for each heatup rate by combining the most restrictive pressure-temperature limits over the complete temperature interval.~~

~~For plant cooldown thermal and pressure stress are additive.~~



RCS Pressure–Temperature Limits for Heatup	Omaha Public Power District Fort Calhoun Station– Unit No. 1	Figure 2–1A

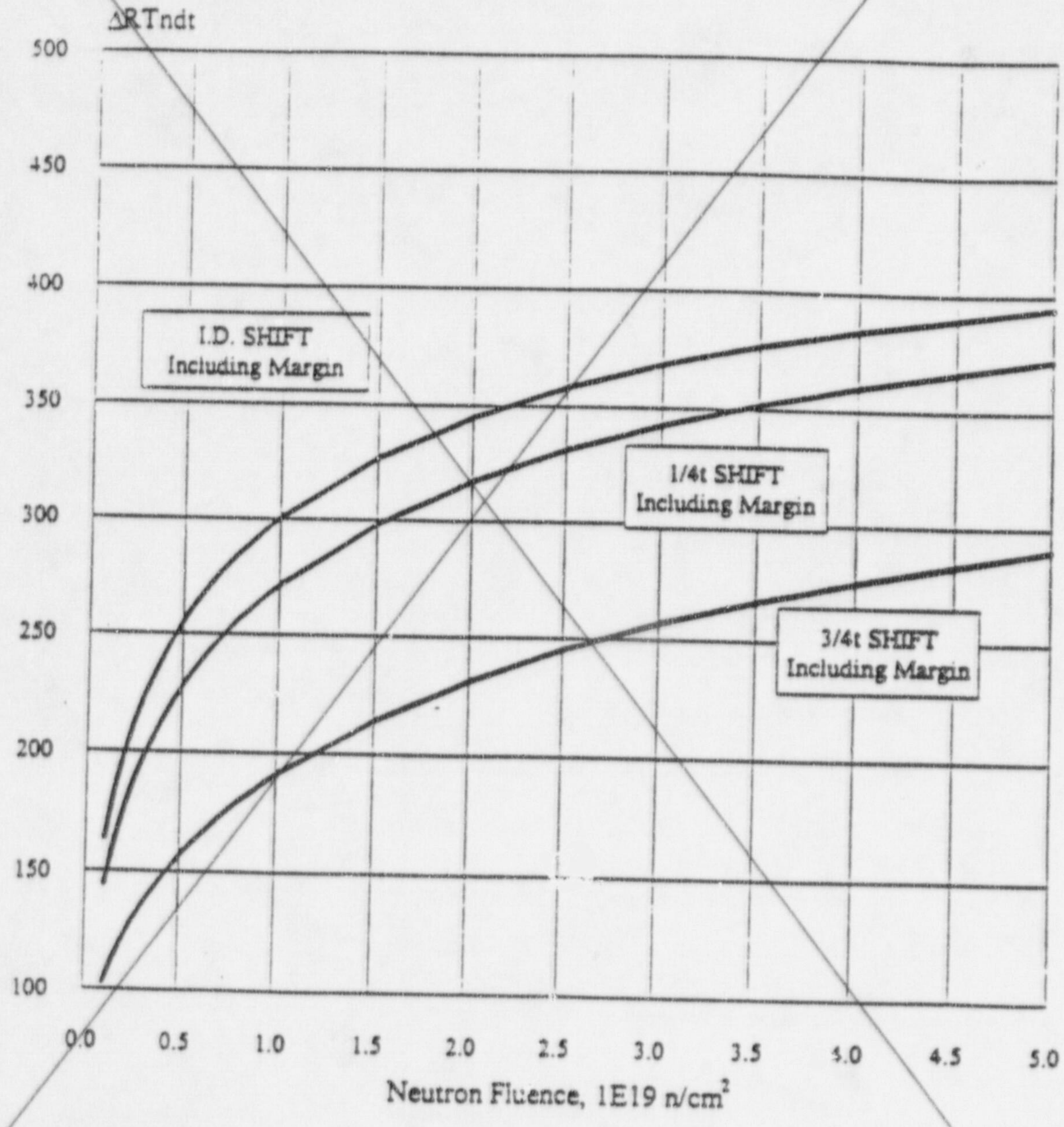
FORT CALHOUN STATION UNIT 1 P/T LIMITS. 20 EFPY
COOLDOWN AND INSERVICE TEST



RCS Pressure–Temperature Limits for Cooldown	Omaha Public Power District Fort Calhoun Station– Unit No. 1	Figure 2–1B

Predicted Radiation Induced NDTT Shift

Fort Calhoun Reactor Vessel Beltline



Predicted Radiation Induced NDTT Shift	Omaha Public Power District Fort Calhoun Station-Unit No. 1	Figure 2-3

Amendment No. 74, 77, 100, 114, 131, 16

2.0 **LIMITING CONDITIONS FOR OPERATION**

2.1 Reactor Coolant System (Continued)

2.1.2 Heatup and Cooldown Curves (Continued)

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

~~M_M = ASME III, Figure G-2214-1~~
~~P = Pressure, psia~~
~~R = Vessel Radius - in.~~
~~t = Vessel Wall Thickness - in.~~

~~$K_{IT} = MT \Delta T_w$~~

~~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 heatup.~~
~~$\frac{M_M R}{t}$ 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. Pressure correction factors for elevation and flow (-56 psia for T_c<210°F and -62 psia for $T_c \geq$210°F) and temperature instrumentation uncertainties (+16°F) are considered when plotting the curves. Pressure instrumentation uncertainty is also considered above the LTOP enable temperature of 385°F. Below this temperature, pressure instrumentation uncertainty is accounted for in the LTOP PORV setpoints.~~

~~B. Inservice Hydrostatic Test - The inservice 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 cannot be exceeded. Taking into account pressure correction factors for elevation and flow, this pressure is (.20)(3125) - 56 = 569 psia, where 56 psi is the hydrostatic head correction factor.~~

~~D. Boltup Temperature = 10°F + 60°F + 12°F = 82°F. At pressure below 569 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. The temperature at which the heatup and cooldown rates change in Figures 2-1A and 2-1B reflects the point at which the most limiting heatup and cooldown rates with respect to the inlet temperature (T_c) change.~~

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.

(9) Fort Calhoun Station Unit No. 1 RCS Pressure - Temperature Limits Report (PTLR)

(10) CE NPSD-683, "Development of a RCS Pressure and Temperature Limits Report for the Removal of P-T Limits and LTOP Requirements from the Technical Specifications," (Latest Approved Revision)

2.0 **LIMITING CONDITIONS FOR OPERATION**
2.1 Reactor Coolant System (continued)
2.1.6 Pressurizer and Main Steam Safety Valves

Applicability

Applies to the status of the pressurizer and main steam safety valves.

Objective

To specify minimum requirements pertaining to the pressurizer and main steam safety valves.

Specifications

To provide adequate overpressure protection for the reactor coolant system and steam system, the following safety valve requirements shall be met:

(1) The reactor shall not be made critical unless the two pressurizer safety valves are operable with their lift settings adjusted to ensure valve opening at 2485 psig ±1% and 2530 psig. ±1%.(1)

(2) Whenever there is fuel in the reactor, and the reactor vessel head is installed, a minimum of one operable safety valve shall be installed on the pressurizer. However, when in at least the cold shutdown condition, safety valve nozzles may be open to containment atmosphere during performance of safety valve tests or maintenance to satisfy this specification.

(3) At least four of the five Main Steam Safety Valves (MSSVs) associated with each steam generator shall be OPERABLE in MODES 1 and 2. Lift settings shall be at 985 psig +3/-2%, 1000 psig +3/-2%, 1010 psig +3/-2%, 1025 psig +3/-2%, and 1035 psig +3/-2%.(1)

a. With less than four of the five MSSVs associated with each steam generator OPERABLE, be in at least HOT STANDBY within 6 hours and HOT SHUTDOWN within an additional 6 hours.

(4) Two power-operated relief valves (PORVs) shall be operable during heatups and cooldowns when the RCS temperature is less than 515°F, and in Modes 4 ~~and 5 whenever the head is on the reactor~~ vessel and the RCS is not vented through a 0.94 square inch or larger vent, to prevent violation of the pressure-temperature limits designated ~~by~~ in the PTLR ~~Figures 2-1A and 2-1B~~.

a. With one PORV inoperable during heatups and cooldowns when the RCS temperature is less than 515°F, restore the inoperable PORV to operable within 7 days or be in cold shutdown within the next 36 hours and depressurize and vent the RCS through at least a 0.94 square inch or larger vent within the following 36 hours.

b. With both PORVs inoperable during heatups and cooldowns when the RCS temperature is less than 515°F, be in cold shutdown within the next 36 hours and depressurize and vent the RCS through at least a 0.94 square inch or larger vent within the following 36 hours.

c. With one PORV inoperable in Modes 4 or 5, within one hour ensure the pressurizer steam space is greater than the minimum volume for RCP startup as specified in the PTLR ~~53% volume (50.6% or less actual level)~~ and restore the inoperable PORV to operable within 7 days. If adequate steam space cannot be established within one hour, then restore the inoperable PORV to operable within 24 hours. If the PORV cannot be restored in the required time, depressurize and vent the RCS through at least a 0.94 square inch or larger vent within the next 36 hours.

2.0 **LIMITING CONDITIONS FOR OPERATION**
2.2 Chemical and Volume Control System
2.2.1 Boric Acid Flow Paths - Shutdown

Applicability

Applies to the operational status of the boric acid flow paths in MODES 4 and 5 when fuel is in the reactor.

Objective

To assure operability of equipment required to add negative reactivity.

Specification

As a minimum, one of the following boric acid flow paths from an OPERABLE borated water source shall be OPERABLE:

a. A flow path from boric acid storage tank CH-11A via either a boric acid transfer pump or a gravity feed connection and a charging pump to the Reactor Coolant System.

b. A flow path from boric acid storage tank CH-11B via either a boric acid transfer pump or a gravity feed connection and a charging pump to the Reactor Coolant System.

c. A flow path from both boric acid storage tanks (CH-11A and CH-11B) via either a boric acid transfer pump or gravity feed connection and a charging pump to the Reactor Coolant System.

d. A flow path from the SIRW tank via either a charging pump or a high pressure safety injection pump to the Reactor Coolant System. The flow path from the SIRW tank to the Reactor Coolant System via a single HPSI pump shall only be established if the requirements in the PTLR are met.

Required Actions

(1) With none of the above boric acid flow paths OPERABLE, suspend all operations involving CORE ALTERATIONS or positive reactivity changes.

2.0 **LIMITING CONDITIONS FOR OPERATION**
2.2 Chemical and Volume Control System (Continued)
2.2.3 Charging Pumps - Shutdown

Applicability

Applies to the operational status of charging pumps in MODES 4 and 5 when fuel is in the reactor.

Objective

To assure operability of equipment required to add negative reactivity.

Specification

At least one charging pump or one high pressure safety injection pump in the boric acid flow path required to be OPERABLE pursuant to Specification 2.2.1 shall be OPERABLE. The flow path from the SIRW tank to the Reactor Coolant System via a single HPSI pump shall only be established if the requirements in the PTLR are met.

Required Actions

(1) With no charging pump or high pressure safety injection pump OPERABLE, suspend all operations involving CORE ALTERATIONS or positive reactivity changes.

2.0 LIMITING CONDITIONS FOR OPERATION

2.2 Chemical and Volume Control System (Continued)

Basis (Continued)

Charging Pumps

Whenever the reactor coolant temperature (T_{cold}) is greater than or equal to 210°F, two charging pumps must be operable in order to ensure it is possible to inject concentrated boric acid into the reactor coolant system with an assumed single failure. With only one pump operable, 72 hours is allowed to restore the system to two operable charging pumps. This is consistent with the allowed outage time for the borated water sources and flow paths required during these modes.

In Modes 4 and 5 when fuel is in the reactor, only one charging pump or high pressure safety injection pump must be operable. This is consistent with the number of operable borated water sources and flow paths required during these modes. A pump is required in order to complete an operable flow path to the reactor coolant system. There are additional restrictions on the use of high pressure safety injection pumps contained in the PTLR ~~Technical Specification 2.3~~ to ensure that the reactor vessel is not overpressurized.

Figure 2-12 contains a 10° F bias to account for temperature measurement uncertainty. An administrative procedure to monitor the temperature of the BASTs and boric acid system piping in the Auxiliary Building ensures that the temperature requirements of Figure 2-12 are met. Should the system temperature be unacceptable for operation at the current boric acid concentration, steps will be taken to reduce the boric acid concentration or raise the temperature of the system such that the concentration is within the acceptable range of Figure 2-12.

The limits on component operability and the time periods for inoperability were selected on the basis of the redundancy indicated above and NUREG-0212 Revision 2. The allowed outage times for the various components are consistent such that a support system has the same allowed outage time as the supported system.

References

(1) USAR Section 9.2

2.0 LIMITING CONDITIONS FOR OPERATION

2.3 Emergency Core Cooling System (Continued)

(3) Protection Against Low Temperature Overpressurization

Operability requirements for HPSI pumps are provided in the RCS Pressure-Temperature Limits Report (PTLR) and shall be adhered to. If not in compliance with the PTLR, actions shall be taken immediately to restore compliance to the PTLR. ~~The following limiting conditions shall be applied during scheduled heatups and cooldowns. Disabling of the HPSI pumps need not be required if the RCS is vented through at least a 0.94 square inch or larger vent.~~

~~Whenever the reactor coolant system cold leg temperature is below 385°F, at least one (1) HPSI pump shall be disabled.~~

~~Whenever the reactor coolant system cold leg temperature is below 320°F, at least two (2) HPSI pumps shall be disabled.~~

~~Whenever the reactor coolant system cold leg temperature is below 270°F, all three (3) HPSI pumps shall be disabled.~~

~~In the event that no charging pumps are operable when the reactor coolant system cold leg temperature is below 270°F, a single HPSI pump may be made operable and utilized for boric acid injection to the core, with flow rate restricted to no greater than 120 gpm.~~

(4) Trisodium Phosphate (TSP) Dodecahydrate

During operating Modes 1 and 2, the TSP baskets shall contain ≥ 110 ft^3 of active TSP.

a. With the above TSP requirements not within limits, the TSP shall be restored within 72 hours.

b. With Specification 2.3(4)a required action and completion time not met, the plant shall be in hot shutdown within the next 6 hours and cold shutdown within the following 36 hours.

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.~~

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~~ DELETED

(7) USAR, Section 4.4.3

3.0 SURVEILLANCE REQUIREMENTS

3.3 Reactor Coolant System and Other Components Subject to ASME XI Boiler & Pressure Vessel Code Inspection and Testing Surveillance

Applicability

Applies to in-service surveillance of primary system components and other components subject to inspection and testing according to ASME XI Boiler & Pressure Vessel Code.

Objective

To ensure the integrity of the reactor coolant system and other components subject to inspection and testing according to ASME XI Boiler & Pressure Vessel Code.

Specifications

(1) Surveillance of the ASME Code Class 1, 2 and 3 systems, except the steam generator tubes inspection, should be covered by ASME XI Boiler & Pressure Vessel Code.

a. In-service inspection of ASME Code Class 1, Class 2, and Class 3 components, including applicable supports, and in-service testing of ASME Code Class 1, Class 2, and Class 3 pumps and valves shall be performed in accordance with Section XI of the ASME Boiler and Pressure Vessel Code, as required by 10 CFR Part 50, Section 50.55a(g), except where specific written relief has been granted by the Commission pursuant to 10 CFR Part 50, Section 50.55a (g)(6)(i).

b. Surveillance of the reactor coolant pump flywheels shall be performed as indicated in Table 3-6.

c. A surveillance program to monitor radiation-induced changes in the mechanical and impact properties of the reactor vessel materials shall be maintained in accordance with 10 CFR Part 50 Appendix H .[1] Examination results shall be used to update the PTLR.

(2) Surveillance of Reactor Coolant System Pressure Isolation Valves

a. Periodic leakage testing* on each valve listed in Table 2-9 shall be accomplished prior to entering the power operation mode every time the plant is placed in the cold shutdown

---

* To satisfy ALARA requirements, leakage may be measured indirectly (as from the performance of pressure indicators) if accomplished in accordance with approved procedures and supported by computations showing that the method is capable of demonstrating valve compliance with the leakage criteria.

New New

5.0 ADMINISTRATIVE CONTROLS

5.9.6 Reactor Coolant System (RCS) Pressure - Temperature Limits Report (PTLR)

a. Reactor Coolant System pressure and temperature limits for heatup, cooldown, low temperature operation, criticality, and hydrostatic testing as well as heatup and cooldown rates shall be established and documented in the PTLR for the following: Technical Specifications 2.1.1, 2.1.2, 2.1.6, 2.2.1, 2.2.3, and 2.3.

b. The analytical methods used to determine the RCS pressure and temperature limits and predicted radiation induced NDTT shift shall be those previously reviewed and approved by the NRC, specifically those described in the following documents:

1. 10 CFR 50.61, "Fracture Toughness Requirements for Protection Against Pressurized Thermal Shock Events"

2. 10 CFR Part 50, Appendix G, "Fracture Toughness Requirements"

3. 10 CFR Part 50, Appendix H, "Reactor Vessel Material Surveillance Program Requirements"

4. Regulatory Guide 1.99, "Radiation Embrittlement of Reactor Vessel Materials," Revision 2, 05/88

5. ASME Boiler and Pressure Vessel Code Section III, Appendix G, "Protection Against Nonductile Failure," 1986 Edition

6. CE NPSD-683, "Development of a RCS Pressure and Temperature Limits Report for the Removal of P-T Limits and LTOP Requirements from the Technical Specifications," (Latest Approved Revision)

7. WCAP-14040-NP-A (Section 2.2; Neutron Fluence Calculation), "Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves," January 1996

c. The PTLR shall be provided to the NRC upon issuance for each reactor vessel fluence period (i.e., the number of EFPY used in the P-T Limit/LTOP analyses) and for any revision or supplement thereto.

New New

ATTACHMENT B

DISCUSSION, JUSTIFICATION AND NO SIGNIFICANT HAZARDS CONSIDERATION

DISCUSSION AND JUSTIFICATION

Omaha Public Power District (OPPD) is proposing revisions to the Fort Calhoun Station Unit No. 1 Technical Specifications (TS) in accordance with Generic Letter (GL) 96-03, "Relocation of the Pressure Temperature Limit Curves and Low Temperature Overpressure Protection System Limits," dated January 31, 1996. The proposed changes are consistent with the recommendations of Combustion Engineering Owners Group (CEOG) Task 942, "Development of a RCS Pressure and Temperature Limits Report for the Removal of P-T Limits and LTOP Requirements from the Technical Specifications," CE NPSD-683. Accordingly, Topical Report CE NPSD-683 is included for NRC approval as a lead CEOG plant submittal. These documents supersede those provided by OPPD's letter (S. K. Gambhir) to the NRC (Document Control Desk) dated January 30, 1998 (LIC-98-0013). The neutron fluence analysis methodology employed is described in Attachment C of OPPD's letter (S. K. Gambhir) to the NRC (Document Control Desk) dated January 30, 1998 (LIC-98-0009). The Fort Calhoun Station neutron fluence analysis was performed by Westinghouse using the methods of WCAP-14040-NP-A and Draft Regulatory Guide-1053, with the ENDF/B-VI Cross-Section Library.

The pressure-temperature (P-T) curves (Figures 2-1A & 2-1B), the predicted radiation induced NDTT shift curve (Figure 2-3) and the low temperature overpressure protection (LTOP) limits (TS 2.3(3)) are proposed for relocation to a document entitled "Fort Calhoun Station Unit No. 1 Reactor Coolant System (RCS) Pressure-Temperature Limits Report (PTLR)." To ensure that the RCS is not over pressurized when the plant is in Mode 4 or Mode 5 with fuel in the reactor, TS 2.2.1.d and TS 2.2.3 are being revised so that PTLR requirements must be met when a flowpath from the safety injection refueling water (SIRW) tank to the RCS is established via a single HPSI pump.

A definition of the RCS PTLR is being added to the TS as well as a new administrative control (TS 5.9.6). TS 5.9.6 establishes the scope of the PTLR, the NRC and ASME approved analytical methods utilized in the PTLR, and requirements for submitting PTLR revisions. Additional administrative revisions are also proposed, which include relocating certain specific values (e.g., minimum pressurizer steam space) to the PTLR, relocating most of the Basis of TS 2.1.2 to the PTLR, relocating statements from the Basis of TS 2.3 concerning reactor startup using the reactor coolant pumps to the PTLR, and adding the PTLR and Topical Report CE NPSD-683 as references in TS 2.1.2.

It should be noted that Topical Report CE NPSD-683 references the use of the ABB/CE codes ROCS/MC. OPPD used the ROCS/MC codes during the time period (i.e., 1990) in which the attached PTLR heatup/cooldown limit curves were generated. However, OPPD has upgraded to the CASMO/SIMULATE codes as contained in Topical Report OPPD-NA-8302-P, Rev. 04, dated May 1994 (Reload Analysis Neutronics Methodology). NRC approval of Topical Report OPPD-NA-8302-P, Rev. 04 is documented in a letter dated December 16, 1994, from S. D. Bloom (NRC) to T. L. Patterson (OPPD). Therefore, future use of the CASMO/SIMULATE codes by OPPD is considered equivalent to the use of the ROCS/MC codes described in Topical Report CE NPSD-683.

DISCUSSION AND JUSTIFICATION (Continued)

As required by GL 96-03, OPPD utilizes NRC approved methodology (ASME Section III, Appendix G) to derive the parameters used to construct the P-T curves and LTOP setpoints. The NDTT shift curve is also derived using NRC approved methodology (Regulatory Guide 1.99, Revision 2).

As described in OPPD's "Application for Amendment of Operating License" (S. K. Gambhir) to the NRC (Document Control Desk) dated June 1, 1992, (LIC-92-157A), ABB-CE performed an analysis for Fort Calhoun Station for P-T limits and LTOP system requirements for continued operation through 20 effective full power years. The P-T limits were calculated to meet the regulations of 10 CFR 50 Appendix A, Design Criterion 14 and Design Criterion 31. The limits were developed using the requirements of 10 CFR 50, Appendix G and ASME Section III, Appendix G. In a letter from the NRC (S. D. Bloom) to OPPD (T. L. Patterson) dated March 23, 1994, the NRC approved the amendment request and issued Amendment 161. Note that OPPD is not requesting an exemption for the use of either ASME Code Case 514 or 636 at this time but may elect to do so in the future. Such a change will be submitted to the NRC for approval if OPPD elects to use one of these code cases.

The proposed amendment will reduce the burden on OPPD and NRC resources by eliminating the necessity of processing an amendment request each time a change is made to P-T limits, NDTT shift curves or LTOP setpoints. Future changes to the PTLR will be made using the latest revision of CE NPSD-683 and will be controlled by the requirements of 10 CFR 50.59 (similar to the Core Operating Limits Report). Thus, future changes to the PTLR will not normally require a license amendment to be effective.

BASIS FOR NO SIGNIFICANT HAZARDS CONSIDERATION:

The proposed changes to the Fort Calhoun Station (FCS) Unit No. 1 Technical Specifications (TS) are in accordance with Generic Letter (GL) 96-03, "Relocation of the Pressure Temperature Limit Curves and Low Temperature Overpressure Protection System Limits," dated January 31, 1996. The proposed changes are also consistent with the recommendations of Combustion Engineering Owners Group (CEOG) Task 942, "Development of a RCS Pressure and Temperature Limits Report for the Removal of P-T Limits and LTOP Requirements from the Technical Specifications," CE NPSD-683. The proposed changes do not involve significant hazards consideration because operation of FCS in accordance with these changes would not:

(1) Involve a significant increase in the probability or consequences of an accident previously evaluated.

The proposed changes relocate the reactor coolant system (RCS) pressure-temperature (P-T) curves, the predicted radiation induced NDTT shift curve and the low temperature overpressure protection (LTOP) limits to the Fort Calhoun Station Unit No. 1 RCS Pressure-Temperature Limits Report (PTLR). Compliance with these curves and limits continues to be required by the Technical Specifications. Changes to the curves and limits will be controlled by TS 5.9.6, and must be in accordance with the NRC and ASME approved methodologies listed there and with 10 CFR 50.59.

The FCS PTLR in combination with the limitations imposed by the TS, will ensure the integrity of the reactor vessel pressure boundary. Therefore, the proposed changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

(2) Create the possibility of a new or different kind of accident from any accident previously evaluated.

There will be no physical alterations to the plant configuration (no new or different equipment is being installed). No changes in operating modes or limits are proposed. The TS retain requirements to maintain the RCS within acceptable operational limits established in accordance with NRC and ASME approved methodologies and assure operability of the LTOP system. As such, the TS will continue to require compliance with the limitations being relocated to the FCS PTLR. Therefore, these proposed changes do not create the possibility of a new or different kind of accident from any previously evaluated.

BASIS FOR NO SIGNIFICANT HAZARDS CONSIDERATION (Continued):

(3) Involve a significant reduction in a margin of safety.

This proposed change to the FCS TS is administrative in nature relocating the P-T curves, NDTT curve, LTOP limits and associated TS requirements to the FCS PTLR in accordance with GL 96-03. Future updates of the FCS PTLR will be conducted under the 10 CFR 50.59 process utilizing NRC and ASME approved methodologies (as described in FCS Unit No. 1 PTLR, Rev. 0 and CEOG Topical Report CE NPSD-683). Therefore, the proposed changes do not involve a significant reduction in a margin of safety.

Based on the above considerations, the proposed amendment does not involve significant hazards considerations as defined by 10 CFR 50.92 and the proposed changes will not result in a condition which significantly alters the impact of the Station on the environment. Thus, the proposed changes meet the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9) and pursuant to 10 CFR 51.22(b) no environmental assessment need be prepared.

ATTACHMENT C