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OPPD
Omaha Public Power District

444 South 16th Street Mall
Omaha NE 68102-2247

July 31, 2003
LIC-03-0104

U. S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, D.C. 20555

References:
1. Docket No. 50-285
2. Letter from OPPD (D. J. Bannister) to NRC (Document Control Desk) dated October 8, 2002, Fort Calhoun Station Unit No. 1 License Amendment Request, "Reactor Coolant System (RCS) Pressure and Temperature Limits Report (PTLR)" (LIC-02-0109)
3. Letter from OPPD (R. T. Ridenoure) to NRC (Document Control Desk) dated April 10, 2003, Response to Request For Additional Information, Pressure-Temperature Limit Report Amendment Request (LIC-03-0052)
4. Letter from OPPD (R. T. Ridenoure) to NRC (Document Control Desk) dated May 20, 2003, Response to Request for Additional Information, Pressure-Temperature Limits Report Amendment Request; Low Temperature Over Pressure (LIC-03-0081)

SUBJECT: **Fort Calhoun Station Unit No. 1 "Technical Specifications Updated for NRC Comments on Pending Amendment Request on the Pressure-Temperature Limits Report"**

In support of the license amendment request, "Reactor Coolant System (RCS) Pressure and Temperature Limits Report (PTLR)" (Reference 2), the Omaha Public Power District (OPPD) provided responses to the Nuclear Regulatory Commission (NRC) Requests for Additional Information in Reference 3 and 4. This transmittal re-submits the proposed Technical Specifications which were originally submitted by Reference 2. The attached proposed Technical Specifications have been updated to include: 1.) The addition of references to the Technical Specification 5.9.6 requested by the NRC during their review of the amendment request, 2.) The update of Technical Specification 5.9.6 reflecting the NRC's approval of OPPD's exemption request, made in Reference 2, for use of the Combustion Engineering methodology presented in CE NPSD-683-A, Revision 6 as basis for the Fort Calhoun PTLR, and 3.) The amendments approved by the NRC following the submittal of Reference 2.

I declare under penalty of perjury that the forgoing is true and correct (Executed on July 31, 2003). No commitments are made to the NRC in this letter.

A001

If you have any questions or require additional information, please contact Dr. R. L. Jaworski of the Fort Calhoun Station Licensing staff at (402) 533-6833.

Sincerely,

R. T. Ridenoure
Division Manager
Nuclear Operations

RTR/RLJ/rlj

Attachments:
1. Marked-Up - Technical Specifications Updated for Pending Amendment of the Pressure-Temperature Limits Report
2. Clean Copy - Technical Specifications Updated for Pending Amendment of the Pressure-Temperature Limits Report

c: T. P. Gwynn, Acting Regional Administrator, NRC Region IV
A. B. Wang, NRC Project Manager
J. G. Kramer, NRC Senior Resident Inspector

Attachment 1

Marked-Up
Technical Specifications
Updated for Pending Amendment
of the
Pressure-Temperature Limits Report

TABLE OF CONTENTS (Continued)

TECHNICAL SPECIFICATIONS - FIGURES

TABLE OF CONTENTS

$\bar{E}$ - 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 PTLR is a fluence dependent document that provides Limiting Conditions for Operation (LCO) in the form of pressure-temperature (P-T) limits to ensure prevention of brittle fracture. In addition, this document establishes power operated relief valve setpoints 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 (EFPYs) specified in the PTLR. NRC approved methodologies are used as the bases for the information provided in the PTLR.

References

(1) USAR, Section 7.2
(2) USAR, Section 7.3

TECHNICAL SPECIFICATIONS

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 pressure and temperature limit Figure(s) shown in the PTLR~~Figure 2-1~~.

(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) Low Temperature Overpressure Protection (LTOP)

(a) The LTOP enable temperature and RCP operations shall be maintained in accordance with the PTLR.

(b) The unit can not be placed on shutdown cooling until the RCS has cooled to an indicated RCS temperature of less than or equal to 300°F.

(c) If no reactor coolant pumps are operating, a non-operating reactor coolant pump shall not be started while T_c is below the LTOP enable temperature stated in the PTLR ~~385°F~~ unless there is a minimum indicated pressurizer steam space of at least 50% by volume.~~at least one of the following conditions is met:~~

 Amendment No. ~~39,56,66,71,119,136,~~
~~161,188~~, 207

2.0 LIMITING CONDITIONS FOR OPERATION

2.1 Reactor Coolant System (Continued)

~~2.1.1 Operable Components (Continued)~~

~~(a) A pressurizer steam space of 53% by volume or greater (50.6% or less actual level) exists, or~~

~~(b) The steam generator secondary side temperature is less than 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.

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 in^2. This requirement may be met by removal of the pressurizer manway which has a cross-sectional area greater than 47 in^2.

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 bases for the LTOP system requirements are documented in the PTLR.~~The LTOP enable temperature 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.~~

TECHNICAL SPECIFICATIONS

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 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 53%, limitation of the steam generator secondary side/RCS cold leg ΔT to 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

TECHNICAL SPECIFICATIONS

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

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 on the pressure and temperature (P-T) limit Figure(s) in the PTLR.~~Figure 2-1.~~

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 on the P-T limit Figure(s) in the PTLR~~Figures 2-1~~.

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

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

Required Actions

(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.

(2) Before the radiation exposure of the reactor vessel exceeds the exposure for which they apply, the P-T limit Figure(s) shown in the PTLR ~~Figure 2-1~~ shall be updated in accordance with the following criteria and procedures:

(a) The P-T limit Figure(s) are ~~Figure 2-1 is~~ valid for a fast neutron (E ≥ 1 MeV) fluence value and corresponding ~~of 2.15X10¹⁹ n/cm² which corresponds to 40~~ EFPY as stated in the PTLR.

(b) The limit line on the P-T limit Figure(s) shown in the PTLR ~~figure~~ 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).

(c) The limit lines in the P-T limit Figure(s) shown in the PTLR ~~Figure 2-1~~ 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 6482°F as it is set by the RT_{NDT} of the reactor vessel flange and is not subjected to a fast neutron flux. The lowest service temperature shall remain at 164°F because components related to this temperature are also not subjected to a fast neutron flux.

(d) ~~The~~ Technical Specification 2.3(3) shall be reviewed and revised as necessary each time the curves on the P-T limit Figure(s) as shown in the PTLR ~~of Figure 2-1~~ 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 limit curves, low temperature overpressure protection system limits, neutron fluence, and adjusted reference temperatures are contained in the PTLR, which was developed using the methodologies stated in Technical Specification 5.9.6(b) and in the PTLR[2].

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 XI[(2)] of the ASME Code including Appendix A and G, Westinghouse Electric Company/Combustion Engineering's P-T (W/CE's) limit curve methodology[(11)] 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 initial 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 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 Reference 5 with the result that the integrated fast neutron flux ($E \geq 1$ MeV) is 1.73×10^{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.~~

~~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 1.73×10^{19} n/cm^2 at the vessel inside surface for 40 years operation at~~

~~1500 MWt and 85% load factor. The predicted shift at this location at the 1/4t depth from the inner surface is projected to be 252°F, including margin, using the shift prediction equation of Regulatory Guide 1.99, Revision 2. The actual shift in~~ $\cancel{T_{NDT}}$ ~~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~~ $\cancel{T_{NDT}}$ ~~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 three removed irradiated reactor vessel surveillance specimens[6,8 and 10], combined with weld chemical composition data and implementation of extreme low radial leakage core loading designs beginning in Cycle 14, indicate that the fluence at the end of 40 Effective Full Power Years (EFPY) at 1500 MWt will be~~ $\cancel{2.15\times10^{19}\ n/cm^2}$ ~~on the inside surface of the reactor vessel. This results in a total shift of the~~ $\cancel{RT_{NDT}}$ ~~of 237.76°F, including margin, for the area of greatest sensitivity (weld metal) at the 1/4t location using Regulatory Guide 1.99, Revision 2, and a shift of 187.97°F at the 3/4t location. Operation through fuel Cycle 34 will result in less than 40 EFPY.~~

~~The limit lines in Figures 2-1 are based on Reference 2, Appendix G, W/CE's methodology for P-T limit curve generation[11], and ASME Code Case N-640 as discussed below.~~

~~Reference Stress Intensity Factor~~

~~The reference stress intensity factor~~ $\cancel{(K_{IR})}$ ~~used in the development of the limit lines in Figure 2-1 is based on ASME Code Case N-640. This Code case allows the use of~~ $\cancel{K_{Ic}}$ ~~(lower bound of static initiation critical stress intensity factor) and is an approved exemption by the NRC in accordance with 10 CFR 50.60(b).~~ $\cancel{K_{Ic}}$ ~~is obtained from a reference fracture toughness curve for reactor pressure vessel low alloy steels as defined in Appendix A to Section XI of the ASME Code and is approximated by the following equation:~~

$$\cancel{K_{Ic} = 33.20 + 20.734e^{[0.0200(T-RT_{NDT})]}}$$

~~where,~~

$\cancel{K_{Ic}}$ = ~~Crack initiation reference stress intensity factor,~~ $\cancel{Ksi\sqrt{in}}$

~~T = temperature at the postulated crack tip, °F~~

$\cancel{RT_{NDT}}$ = ~~adjusted reference nil ductility temperature (also called ART) at the postulated crack tip, °F~~

~~For any instant during the postulated heatup or cooldown,~~ $\cancel{K_{Ic}}$ ~~is calculated using the metal temperature at the tip of the flaw, as well as the value of ART at that flaw location.~~

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

Regulatory Requirements

The Reference 2, Appendix G equation relating K_{IM}, K_{IT}, and K_{IR} is re-arranged as shown below to solve for the allowable pressure stress intensity factor, K_{IM}, as a function of time with the calculated K_{IR} and K_{IT} values to determine the allowable pressure stress intensity factor and consequently the allowable pressure:

(1) For Service Level A and B operation:

$$K_{IM} = \frac{K_{IR} - K_{IT}}{2}$$

and

(2) For Hydrostatic and Test Conditions when the core is not critical and tests are performed at isothermal conditions (i.e. thermal stress intensity factor, K_{IT}, = 0)

$$K_{IM} = \frac{K_{IR}}{1.5}$$

where,

K_{IM} = Allowable pressure stress intensity factor based on coolant temperature, $Ksi\sqrt{in}$

K_{IR} = Reference stress intensity factor based on coolant temperature, $Ksi\sqrt{in}$

K_{IT} = Thermal stress intensity factor based on coolant temperature, $Ksi\sqrt{in}$

Calculational of P-Allowable

To develop P-T limits, the reactor vessel (RV) beltline region is the only location that receives sufficient neutron fluence to substantially alter the fracture toughness of the RV material. Hence, the beltline region is the most limiting with respect to allowable pressure at any specific temperature. This reduction in fracture toughness is determined using an adjusted reference temperature (ART), which is calculated in accordance with Regulatory Guide 1.99 Revision 2. The allowable pressure is based on the highest ART. The RV beltline region is analyzed assuming a semi-elliptical surface flaw oriented in the axial direction with a depth of one quarter of the RV beltline thickness and an aspect ratio of one to six.

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

This postulated flaw is analyzed at both the inside diameter location (referred to as the 1/4t location) and the outside diameter location (referred to as the 3/4t location) to assure the most limiting condition is achieved. K_{IT} is determined in accordance with Reference 2, Appendix G-2214.3. The thermal stress intensity factors are determined from the calculated temperature profile through the beltline wall using thermal influence coefficients specifically generated for this purpose. The method employed uses a polynomial fit of the temperature profile and superposition using influence coefficients to calculate K_{IT}. The influence coefficients are dependent upon the geometrical parameters associated with the postulated defect, the geometry of the reactor vessel beltline region, and the assumed unit loading. These influence coefficients were calculated using a 2-dimensional finite element model of the reactor vessel. The influence coefficients were corrected for three-dimensional effects using Reference 2, Appendix A procedures. The K_{IT} and K_{IM} are calculated at any time point in a transient using influence coefficients generated by applying unit loads on a finite element model of the reactor vessel beltline region. The influence coefficients are calculations of stress intensity factors at the 1/4t and 3/4t crack depth location under the following unit loads:

a. for K_{IM-P}, pressure load of 1 ksi, $ksi\text{-}\sqrt{in}/ksi$

b. for K_{IT-L}, linear through-wall gradient with peak temperature of 1°F, $ksi\text{-}\sqrt{in}/°F$

c. for K_{IT-Q}, quadratic through-wall gradient with peak temperature of 1°F, $ksi\text{-}\sqrt{in}/°F$

d. for K_{IT-C}, cubic through-wall gradient with peak temperature of 1°F, $ksi\text{-}\sqrt{in}/°F$

Each stress intensity factor is calculated using a standard quarter point element formulation at the respective crack tips. Since all calculations performed are linear, superposition is then used to scale and combine these influence coefficients as necessary to determine the stress intensity factor for a given temperature profile.

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

In the case of K_{IT}, the through-wall temperature profiles are then fit to the third order polynomial below:

$$T(x) = C_o + C_L(1-X/_h) + C_Q(1-X/_h)2 + C_C(1-X/_h)3$$

where,

T(x)	=	Temperature at radial location x from inside wall surface, °F
C_o, C_L, C_Q, C_C	=	Coefficients in polynomial fit, °F
x	=	Distance through beltline wall, in
h	=	Beltline wall thickness, in

The coefficients of this polynomial are then combined through the following equation to calculate K_{IT} at the 1/4t and 3/4t locations.

$$K_{IT-Total} = C_L * K_{IT-L} + C_Q * K_{IT-Q} + C_C * K_{IT-C}$$

To calculate the allowable pressure, P-Allowable, the resultant $K_{IT-Total}$ from above in conjunction with Equation (1) from Reference 2, Appendix G-2215, is described as follows.

for Normal Level A and B loads

$$P\text{-}Allowable = \frac{K_{IR} - K_{IT-Total}}{2 * K_{IM-P}}$$

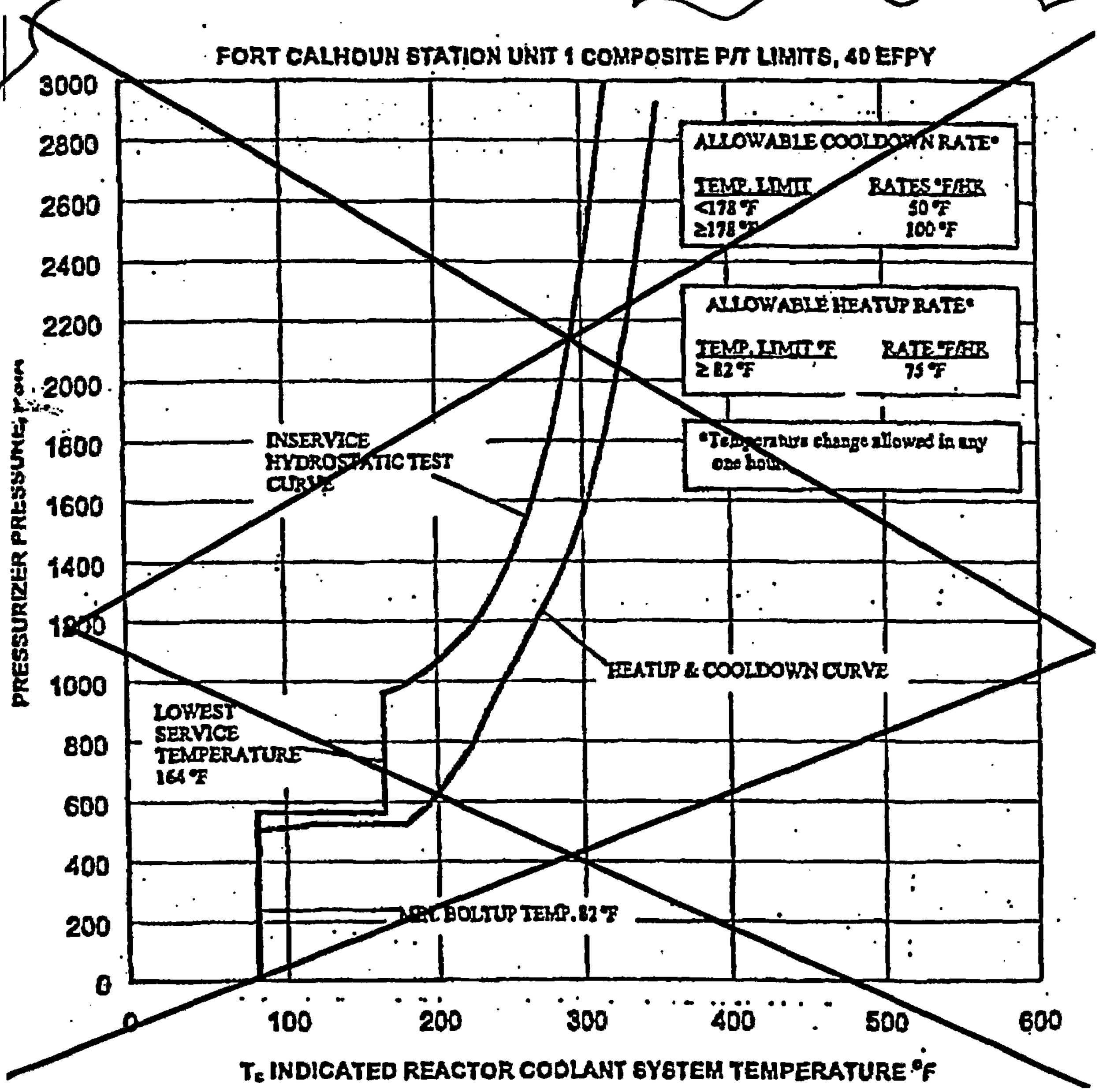
for Hydrostatic and Test Conditions, where Isothermal conditions result in $K_{IT-Total} = 0$

$$P\text{-}Allowable = \frac{K_{IR}}{1.5 * K_{IM-P}}$$

The P-T limits developed using the method described above account for the temperature differential between the RV base metal and the reactor coolant bulk fluid temperature only. However, uncertainties for instrumentation error, elevation, and flow induced differential pressure differences between the RV beltline and pressurizer are accounted for as follows:

1) Temperature instrumentation uncertainty of 14°F is applied to the entire temperature range of Figure 2-1.

~~Figure 2-1~~



~~RCS Pressure-Temperature Limits for Heatup, Cooldown, and Inservice Test~~	~~Omaha Public Power District Fort Calhoun Station Unit No. 1~~

~~2) Pressure correction factors that account for the difference in pressure between the reactor vessel beltline and pressurizer pressure instrumentation due to elevation and RCP flow are as follows:~~

~~RCS Temperature < 210°F = 61 psi~~
~~RCS Temperature ≥ 210°F = 67 psi~~

~~3) Below 350°F, pressure instrumentation uncertainty is accounted for in the LTOP system setpoints. Above 350°F, a pressure instrumentation uncertainty of 50 psi is applied to Figure 2-1.~~

~~Lowest Service Temperature = 50°F + 100°F + 14°F = 164°F. As indicated previously, an RT_{NDT} for all material with the exception of the reactor vessel beltline was established at 50°F. Reference 2, Section III, NB-2332 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 cannot be exceeded. Taking into account pressure correction factors for elevation and flow, this pressure is (.20)(3125) - 61 = 564 psia, where 61 psi is the pressure correction factor.~~

~~Boltup Temperature = 10°F + 14°F = 24°F. A conservative value of 82°F will be used and maintained. At pressure below 564 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 RT_{NDT} methods. This temperature corresponds to the measured 10°F RT_{NDT} of the reactor vessel flange, which is not subject to radiation damage, plus 14°F instrument error.~~

~~The temperature at which the heatup and cooldown rates change in Figure 2-1 reflects the point at which the most limiting heatup and cooldown rates with respect to the inlet temperature (T_c) change.~~

TECHNICAL SPECIFICATIONS

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

References:

(1) USAR, Section 4.2.2

(2) Technical Data Book IX, Fort Calhoun Station Unit No. 1, RCS Pressure-Temperature Limits Report

~~(2) ASME Boiler and Pressure Vessel Code~~

~~(3) USAR, Section 4.2.4~~

~~(4) USAR, Section 3.4.6~~

~~(5) WCAP-15443, Revision 0, Fast Neutron Fluence Evaluation for the Fort Calhoun Unit 1 Reactor Pressure Vessel, July 2000.~~

~~(6) Technical Specification 2.3(3)~~

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

~~(8) TR-O-MCM-001, Revision 1, Omaha Public Power District, Fort Calhoun Station Unit No. 1, Evaluation of Irradiated Capsule W-225, August 1980.~~

~~(9) TR-O-MCM-002, Omaha Public Power District, Fort Calhoun Station Unit No. 1, Evaluation of Irradiated Capsule W-265, March 1984.~~

~~(10) BAW-2226, Omaha Public Power District, Fort Calhoun Station Unit No. 1, Evaluation of Irradiated Capsule W-275, November 1994.~~

~~(11) Safety Evaluation of Topical Report CE NPSD-683, Revision 6, "Development of a RCS Pressure and Temperature Limits Report (PTLR) for the Removal of P-T Limits and LTOP Requirements from the Technical Specifications" (TAC No. MA9561).~~

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 the P-T limit Figure(s) shown in the PTLR~~Figure 2-1~~.

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 ~~50~~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.

TECHNICAL SPECIFICATIONS

2.0 LIMITING CONDITIONS FOR OPERATION
2.3 Emergency Core Cooling System

Applicability

Applies to the operating status of the emergency core cooling system.

Objective

To assure operability of equipment required to remove decay heat from the core.

Specifications

(1) Minimum Requirements

The reactor shall not be made critical unless all of the following conditions are met:

a. The SIRW tank contains not less than 283,000 gallons of water with a boron concentration of at least the refueling boron concentration at a temperature not less than 50°F.

b. One means of temperature indication (local) of the SIRW tank is operable.

c. All four safety injection tanks are operable and pressurized to at least 240 psig and a maximum of 275 psig with a tank level of at least 116.2 inches (67%) and a maximum level of 128.1 inches (74%) with refueling boron concentration.

d. One level and one pressure instrument is operable on each safety injection tank.

e. One low-pressure safety injection train is operable on each associated 4,160 V engineered safety feature bus. |

f. One high-pressure safety injection pump is operable on each associated 4,160 V engineered safety feature bus.

g. Both shutdown heat exchangers are operable.

h. Piping and valves shall be operable to provide two flow paths from the SIRW tank to the reactor coolant system.

i. All valves, piping and interlocks associated with the above components and required to function during accident conditions are operable. HCV-2914, 2934, 2974, and 2954 shall have power removed from the motor operators by locking open the circuit breakers in the power supply lines to the valve motor operators. FCV-326 shall be locked open.

2.0 LIMITING CONDITIONS FOR OPERATION

2.3 Emergency Core Cooling System (Continued)

(3) Protection Against Low Temperature Overpressurization

The following limiting conditions shall be applied during scheduled heatups and cooldowns. Disabling of the HPSI pumps need not be required if the RCS is vented through at least a 0.94 square inch or larger vent.

Whenever the reactor coolant system cold leg temperature is below ~~350~~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 ≥ 126 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. 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)

Components in excess of those allowed by Conditions a, b, d, and e may be inoperable provided they are returned to operable status within 1 hour when performing the quarterly recirculation actuation logic channel functional test (Table 3-2 item 20) under administrative controls. This allowance applies only to the remaining portion of Cycle 20 and all of Cycle 21. This prevents violating Technical Specifications or necessitating a unit shutdown due to inability to perform the quarterly recirculation actuation logic channel functional test. These administrative controls consist of stationing three dedicated operators at the Engineered Safeguards Features (ESF) panel controls in the control room. In this way, the following conditions are maintained and actions can be rapidly performed should a valid ESF actuation occur:

the appropriate Safety Injection Refueling Water Tank (SIRWT) to Safety Injection (SI) and Containment Spray (CS) pumps suction valve control switch is maintained in the OPEN position (spring-return switch),
the appropriate SI and CS pumps to SIRWT recirculation minimum flow valve control switch is maintained in the OPEN position (spring-return switch),
the appropriate Recirculation Actuation Signal (RAS) lockout relays and initiating signal can be rapidly reset,
the appropriate SI and CS pumps to SIRWT recirculation minimum flow valve control switch can be rapidly returned to the AUTO position,
the appropriate SIRWT to SI and CS pumps suction valve control switch can be rapidly returned to the AUTO position, and
the appropriate Containment Sump to SI and CS pumps suction valve control switch can be rapidly returned to the AUTO position.

The appropriate SI and CS pumps to SIRWT recirculation minimum flow valve control switch and the appropriate SIRWT to SI and CS pumps suction valve control switch are held in the OPEN position during the test to enhance the reliability of the appropriate SI and CS pumps by maintaining the associated valves open.

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) ~~Deleted Technical Specification 2.1.2, Figure 2-1B~~
(7) USAR, Section 4.4.3
(8) CE NPSD-994, "CEOG Joint Applications Report for Safety Injection Tank AOT/SIT Extension," May 1995.
(9) CE NPSD-995, "CEOG Joint Applications Report for Low Pressure Safety Injection System AOT Extension," May 1995.

2.0 LIMITING CONDITIONS FOR OPERATION

2.10 Reactor Core

2.10.1 Minimum Conditions for Criticality

Applicability

Applies to the status of the reactor coolant system during reactor criticality.

Objective

To prevent unanticipated power excursions of an unsafe magnitude.

Specifications

(1) ~~Except during physics tests at less than 10⁻⁴% power,~~ The reactor shall not be made critical if the average reactor coolant temperature is below 515°F.

~~(2) In no case shall the reactor be made critical if the reactor coolant temperature is below NDTT +120°F.~~

(2) No more than one CEA at a time in a regulating or non-trippable group shall be exercised or withdrawn until after a steam bubble and normal water level as given in operating procedures are established in the pressurizer.

(3) Reactor coolant boron concentration shall not be reduced below that required for the Hot Shutdown Condition until after a steam bubble and normal water level are established in the pressurizer.

Basis

At the beginning of each fuel cycle, the moderator temperature coefficient is expected to be slightly negative at operating temperatures with all CEA's withdrawn. However, variations in cycle core loading and the uncertainty of the calculation are such that it is possible that a slightly positive coefficient could exist.

The moderator temperature coefficient at lower temperatures will be less negative or more positive than at operating temperature. It is, therefore, prudent to restrict the operation of the reactor when reactor coolant temperatures are less than 515°F.

2.0 LIMITING CONDITIONS FOR OPERATION
2.10 Reactor Core (Continued)

2.10.1 Minimum Conditions for Criticality (Continued)

If the shutdown margin required by the Hot Shutdown Condition is maintained, there is no possibility of an accidental criticality as a result of a change of moderator temperature or a decrease of coolant pressure. Normal water level is established in the pressurizer prior to the withdrawal of CEA or the dilution of boron so as to preclude the possible overpressurization of a solid reactor coolant system.

During physics tests, special operating precautions will be taken. In addition, the strong negative effect of the Doppler coefficient would limit the magnitude of a power excursion resulting from a reduction of moderator density.

~~The requirement that the reactor not be made critical below NDTT +120°F provides increased assurance that the proper relationship between reactor coolant pressure and temperature will be maintained relative to the NDTT of the reactor coolant system. Heatup to this temperature will be accomplished by operating the reactor coolant pumps.~~

TABLE 3-5 (Continued)

		Test	Frequency
19.	Refueling Water Level	Verify refueling water level is ≥ 23 ft. above the top of the reactor vessel flange.	Prior to commencing, and daily during CORE ALTERATIONS and/or REFUELING OPERATIONS inside containment.
20.	Spent Fuel Pool Level	Verify spent fuel pool water level is ≥ 23 ft. above the top of irradiated fuel assemblies seated in the storage racks.	Prior to commencing, and weekly during REFUELING OPERATIONS in the spent fuel pool.
21.	Containment Penetrations	Verify each required containment penetration is in the required status.	Prior to commencing, and weekly during CORE ALTERATIONS and/or REFUELING OPERATIONS in containment.
22.	Spent Fuel Assembly Storage	Verify by administrative means that initial enrichment and burnup of the fuel assembly is in accordance with Figure 2-10.	Prior to storing the fuel assembly in Region 2 (including peripheral cells).
23.	P-T Limit Curve	Verify RCS Pressure, RCS temperature, and RCS heatup and cooldown rates are within the limits specified by the P-T limit Figure(s) shown in the PTLR.	This test is only required during RCS heatup and cooldown operations and RCS inservice leak and hydrostatic testing. While these operations are occurring, this test shall be performed every 30 minutes.

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) Examinations 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.

5.0 ADMINISTRATIVE CONTROLS

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

a. RCS pressure and temperature limits for heatup, cooldown, low temperature overpressure protection, criticality, and hydrostatic testing as well as heatup and cooldown rates shall be established and documented in the PTLR for Technical Specifications 2.1.1 and 2.1.2.

b. The analytical methods used in the PTLR shall be those previously reviewed and approved by the NRC, specifically those described in the following documents:

1. CE NPSD-683-A, Revision 6, "Development of a RCS Pressure and Temperature Limits Report for the Removal of P-T Limits and LTOP Requirements from the Technical Specifications," April 2001.
2. WCAP-15443, Revision 0, "Fast Neutron Fluence Evaluations for the Fort Calhoun Unit 1 Reactor Pressure Vessel," July 2000.
3. Safety Evaluation by the Office of Nuclear Reactor Regulation Related to Amendment Number 199 to Facility Operating License DPR-40 Omaha Public Power District Fort Calhoun Station, Unit Number 1, dated June 7, 2001.
4. CEN-636, Revision 2, "Evaluation of Reactor Vessel Surveillance Data Pertinent to the Fort Calhoun Reactor Vessel Beltline Materials, dated July 2000.
5. FC06876, Revision 0, "Performance of Low Temperature Overpressure Protection System Analyses Using RELAP5: Methodology Paper."
6. FC06877, "Low Temperature Overpressure Protection (LTOP) Analysis, Revision 1."
7. Safety Evaluation by the Office of Nuclear Reactor Regulation Related to Amendment Number 207 to Facility Operating License Number DPR-40 Omaha Public Power District Fort Calhoun Station, Unit Number 1, dated April 22, 2002.
8. Letter LTR-CI-01-25, Revision 0 from Westinghouse Electric Company (S.T. Byrne) to OPPD (J. Jensen), "Assessment of Extended Beltline Limit for Fort Calhoun Station Reactor Pressure Vessel," dated December 18, 2001.
9. WCAP-15741, Revision 0, "Reactor Vessel Surveillance Program Withdrawal Schedule Modifications," dated September 2001.
10. Letter from NRC (A. B. Wang) to Omaha Public Power District (R. T. Ridenoure), Fort Calhoun Station - Unit 1, Exemption from the Requirements of Appendix G to 10 CFR Part 50 (TAC No. MB8237), dated July 30, 2003.

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 analysis) and for any revision or supplement thereto.

Attachment 2

Clean Copy
Technical Specifications
Updated for Pending Amendment
of the
Pressure-Temperature Limits Report

TABLE OF CONTENTS (Continued)

TECHNICAL SPECIFICATIONS - FIGURES

TABLE OF CONTENTS

$\bar{E}$ - 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 PTLR is a fluence dependent document that provides Limiting Conditions for Operation (LCO) in the form of pressure-temperature (P-T) limits to ensure prevention of brittle fracture. In addition, this document establishes power operated relief valve setpoints 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 (EFPYs) specified in the PTLR. NRC approved methodologies are used as the bases for the information 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 pressure and temperature limit Figure(s) shown in the PTLR.

(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) Low Temperature Overpressure Protection (LTOP)

(a) The LTOP enable temperature and RCP operations shall be maintained in accordance with the PTLR.

(b) The unit can not be placed on shutdown cooling until the RCS has cooled to an indicated RCS temperature of less than or equal to 300°F.

(c) If no reactor coolant pumps are operating, a non-operating reactor coolant pump shall not be started while T_c is below the LTOP enable temperature stated in the PTLR unless there is a minimum indicated pressurizer steam space of at least 50% by volume.

(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.

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 in^2. This requirement may be met by removal of the pressurizer manway which has a cross-sectional area greater than 47 in^2.

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 bases for the LTOP system requirements are documented in the PTLR. |

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.

References

(1) USAR Section 4.3.7

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 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:

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 on the pressure and temperature (P-T) limit Figure(s) in the PTLR.

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 on the P-T limit Figure(s) in the PTLR.

c. The heatup rate of the pressurizer shall not exceed l00°F in any one hour period.

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

Required Actions

(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.

(2) Before the radiation exposure of the reactor vessel exceeds the exposure for which they apply, the P-T limit Figure(s) shown in the PTLR shall be updated in accordance with the following criteria and procedures:

(a) The P-T limit Figure(s) are valid for a fast neutron (E ≥ 1 MeV) fluence value and corresponding EFPY as stated in the PTLR.

(b) The limit line on the P-T limit Figure(s) shown in the PTLR 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).

(c) The limit lines in the P-T limit Figure(s) shown in the PTLR 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 64°F as it is set by the RT_{NDT} of the reactor vessel flange and is not subjected to a fast neutron flux. The lowest service temperature shall remain at 164°F because components related to this temperature are also not subjected to a fast neutron flux.

(d) Technical Specification 2.3(3) shall be reviewed and revised as necessary each time the curves on the P-T limit Figure(s) as shown in the PTLR 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 limit curves, low temperature overpressure protection system limits, neutron fluence, and adjusted reference temperatures are contained in the PTLR, which was developed using the methodologies stated in Technical Specification 5.9.6(b) and in the PTLR[2].

TECHNICAL SPECIFICATIONS

2.0 LIMITING CONDITIONS FOR OPERATION

2.1 Reactor Coolant System (Continued)

2.1.2 Heatup and Cooldown Rate (Continued)

References:

(1) USAR, Section 4.2.2

(2) Technical Data Book IX, Fort Calhoun Station Unit No. 1, RCS Pressure-Temperature Limits Report

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 the P-T limit Figure(s) shown in the PTLR. |

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 50% volume 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
2.3 Emergency Core Cooling System

Applicability

Applies to the operating status of the emergency core cooling system.

Objective

To assure operability of equipment required to remove decay heat from the core.

Specifications

(1) Minimum Requirements

The reactor shall not be made critical unless all of the following conditions are met:

a. The SIRW tank contains not less than 283,000 gallons of water with a boron concentration of at least the refueling boron concentration at a temperature not less than 50°F.

b. One means of temperature indication (local) of the SIRW tank is operable.

c. All four safety injection tanks are operable and pressurized to at least 240 psig and a maximum of 275 psig with a tank level of at least 116.2 inches (67%) and a maximum level of 128.1 inches (74%) with refueling boron concentration.

d. One level and one pressure instrument is operable on each safety injection tank.

e. One low-pressure safety injection train is operable on each associated 4,160 V engineered safety feature bus.

f. One high-pressure safety injection pump is operable on each associated 4,160 V engineered safety feature bus.

g. Both shutdown heat exchangers are operable.

h. Piping and valves shall be operable to provide two flow paths from the SIRW tank to the reactor coolant system.

i. All valves, piping and interlocks associated with the above components and required to function during accident conditions are operable. HCV-2914, 2934, 2974, and 2954 shall have power removed from the motor operators by locking open the circuit breakers in the power supply lines to the valve motor operators. FCV-326 shall be locked open.

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

(3) Protection Against Low Temperature Overpressurization

The following limiting conditions shall be applied during scheduled heatups and cooldowns. Disabling of the HPSI pumps need not be required if the RCS is vented through at least a 0.94 square inch or larger vent.

Whenever the reactor coolant system cold leg temperature is below 350°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 ≥ 126 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. 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)

Components in excess of those allowed by Conditions a, b, d, and e may be inoperable provided they are returned to operable status within 1 hour when performing the quarterly recirculation actuation logic channel functional test (Table 3-2 item 20) under administrative controls. This allowance applies only to the remaining portion of Cycle 20 and all of Cycle 21. This prevents violating Technical Specifications or necessitating a unit shutdown due to inability to perform the quarterly recirculation actuation logic channel functional test. These administrative controls consist of stationing three dedicated operators at the Engineered Safeguards Features (ESF) panel controls in the control room. In this way, the following conditions are maintained and actions can be rapidly performed should a valid ESF actuation occur:

the appropriate Safety Injection Refueling Water Tank (SIRWT) to Safety Injection (SI) and Containment Spray (CS) pumps suction valve control switch is maintained in the OPEN position (spring-return switch),
the appropriate SI and CS pumps to SIRWT recirculation minimum flow valve control switch is maintained in the OPEN position (spring-return switch),
the appropriate Recirculation Actuation Signal (RAS) lockout relays and initiating signal can be rapidly reset,
the appropriate SI and CS pumps to SIRWT recirculation minimum flow valve control switch can be rapidly returned to the AUTO position,
the appropriate SIRWT to SI and CS pumps suction valve control switch can be rapidly returned to the AUTO position, and
the appropriate Containment Sump to SI and CS pumps suction valve control switch can be rapidly returned to the AUTO position.

The appropriate SI and CS pumps to SIRWT recirculation minimum flow valve control switch and the appropriate SIRWT to SI and CS pumps suction valve control switch are held in the OPEN position during the test to enhance the reliability of the appropriate SI and CS pumps by maintaining the associated valves open.

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) Deleted
(7) USAR, Section 4.4.3
(8) CE NPSD-994, "CEOG Joint Applications Report for Safety Injection Tank AOT/SIT Extension," May 1995.
(9) CE NPSD-995, "CEOG Joint Applications Report for Low Pressure Safety Injection System AOT Extension," May 1995.

2.0 LIMITING CONDITIONS FOR OPERATION

2.10 Reactor Core

2.10.1 Minimum Conditions for Criticality

Applicability

Applies to the status of the reactor coolant system during reactor criticality.

Objective

To prevent unanticipated power excursions of an unsafe magnitude.

Specifications

(1) The reactor shall not be made critical if the average reactor coolant temperature is below 515°F.

(2) No more than one CEA at a time in a regulating or non-trippable group shall be exercised or withdrawn until after a steam bubble and normal water level as given in operating procedures are established in the pressurizer.

(3) Reactor coolant boron concentration shall not be reduced below that required for the Hot Shutdown Condition until after a steam bubble and normal water level are established in the pressurizer.

Basis

At the beginning of each fuel cycle, the moderator temperature coefficient is expected to be slightly negative at operating temperatures with all CEA's withdrawn. However, variations in cycle core loading and the uncertainty of the calculation are such that it is possible that a slightly positive coefficient could exist.

The moderator temperature coefficient at lower temperatures will be less negative or more positive than at operating temperature. It is, therefore, prudent to restrict the operation of the reactor when reactor coolant temperatures are less than 515°F.

2.0 LIMITING CONDITIONS FOR OPERATION

2.10 Reactor Core (Continued)

2.10.1 Minimum Conditions for Criticality (Continued)

If the shutdown margin required by the Hot Shutdown Condition is maintained, there is no possibility of an accidental criticality as a result of a change of moderator temperature or a decrease of coolant pressure. Normal water level is established in the pressurizer prior to the withdrawal of CEA or the dilution of boron so as to preclude the possible overpressurization of a solid reactor coolant system.

During physics tests, special operating precautions will be taken. In addition, the strong negative effect of the Doppler coefficient would limit the magnitude of a power excursion resulting from a reduction of moderator density.

TABLE 3-5 (Continued)

		Test	**Frequency**
19.	Refueling Water Level	Verify refueling water level is ≥ 23 ft. above the top of the reactor vessel flange.	Prior to commencing, and daily during CORE ALTERATIONS and/or REFUELING OPERATIONS inside containment.
20.	Spent Fuel Pool Level	Verify spent fuel pool water level is ≥ 23 ft. above the top of irradiated fuel assemblies seated in the storage racks.	Prior to commencing, and weekly during REFUELING OPERATIONS in the spent fuel pool.
21.	Containment Penetrations	Verify each required containment penetration is in the required status.	Prior to commencing, and weekly during CORE ALTERATIONS and/or REFUELING OPERATIONS in containment.
22.	Spent Fuel Assembly Storage	Verify by administrative means that initial enrichment and burnup of the fuel assembly is in accordance with Figure 2-10.	Prior to storing the fuel assembly in Region 2 (including peripheral cells).
23.	P-T Limit Curve	Verify RCS Pressure, RCS temperature, and RCS heatup and cooldown rates are within the limits specified by the P-T limit Figure(s) shown in the PTLR.	This test is only required during RCS heatup and cooldown operations and RCS inservice leak and hydrostatic testing. While these operations are occurring, this test shall be performed every 30 minutes.

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) Examinations 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.

5.0 ADMINISTRATIVE CONTROLS

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

a. RCS pressure and temperature limits for heatup, cooldown, low temperature overpressure protection, criticality, and hydrostatic testing as well as heatup and cooldown rates shall be established and documented in the PTLR for Technical Specifications 2.1.1 and 2.1.2.

b. The analytical methods used in the PTLR shall be those previously reviewed and approved by the NRC, specifically those described in the following documents:

1. CE NPSD-683-A, Revision 6, "Development of a RCS Pressure and Temperature Limits Report for the Removal of P-T Limits and LTOP Requirements from the Technical Specifications," April 2001.
2. WCAP-15443, Revision 0, "Fast Neutron Fluence Evaluations for the Fort Calhoun Unit 1 Reactor Pressure Vessel," July 2000.
3. Safety Evaluation by the Office of Nuclear Reactor Regulation Related to Amendment Number 199 to Facility Operating License DPR-40 Omaha Public Power District Fort Calhoun Station, Unit Number 1, dated June 7, 2001.
4. CEN-636, Revision 2, "Evaluation of Reactor Vessel Surveillance Data Pertinent to the Fort Calhoun Reactor Vessel Beltline Materials, dated July 2000.
5. FC06876, Revision 0, "Performance of Low Temperature Overpressure Protection System Analyses Using RELAP5: Methodology Paper."
6. FC06877, "Low Temperature Overpressure Protection (LTOP) Analysis, Revision 1."
7. Safety Evaluation by the Office of Nuclear Reactor Regulation Related to Amendment Number 207 to Facility Operating License Number DPR-40 Omaha Public Power District Fort Calhoun Station, Unit Number 1, dated April 22, 2002.
8. Letter LTR-CI-01-25, Revision 0 from Westinghouse Electric Company (S.T. Byrne) to OPPD (J. Jensen), "Assessment of Extended Beltline Limit for Fort Calhoun Station Reactor Pressure Vessel," dated December 18, 2001.
9. WCAP-15741, Revision 0, "Reactor Vessel Surveillance Program Withdrawal Schedule Modifications," dated September 2001.
10. Letter from NRC (A. B. Wang) to Omaha Public Power District (R. T. Ridenoure), Fort Calhoun Station - Unit 1, Exemption from the Requirements of Appendix G to 10 CFR Part 50 (TAC No. MB8237), dated July 30, 2003.

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 analysis) and for any revision or supplement thereto.