UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

OMAHA PUBLIC POWER DISTRICT

DOCKET NO. 50-285

FORT CALHOUN STATION, UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 161
License No. DPR-40

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by the Omaha Public Power District (the licensee) dated December 19, 1990, and June 1, 1992 as supplemented by letters dated February 1, 1993, and February 25, 1994, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance: (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this license amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, Facility Operating License No. DPR-40 is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 3.B. of Facility Operating License No. DPR-40 is hereby amended to read as follows:

   B. Technical Specifications

   The Technical Specifications contained in Appendix A, as revised through Amendment No. **161** , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. The license amendment is effective as of its date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Paul W. Johnson

for William D. Beckner, Director
Project Directorate IV-1
Division of Reactor Projects - III/IV/V
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: **March 23, 1994**

ATTACHMENT TO LICENSE AMENDMENT NO. 161

FACILITY OPERATING LICENSE NO. DPR-40

DOCKET NO. 50-285

Revise Appendix "A" Technical Specifications as indicated below. The revised pages are identified by amendment number and contain vertical lines indicating the area of change.

REMOVE PAGES	INSERT PAGES
viii	viii
2-2a	2-2a
2-2b	2-2b
2-2c	2-2c
2-2d	2-2d
2-3	2-3
2-4	2-4
2-6	2-6
Figure 2-1A	Figure 2-1A
Figure 2-1B	Figure 2-1B
Figure 2-3	Figure 2-3
2-7	2-7
2-7a	2-7a
2-15	2-15
2-15a	2-15a
2-16	2-16
2-22	2-22
2-23a	2-23a
3-16a	3-16a

TECHNICAL SPECIFICATIONS - FIGURES

TABLE OF CONTENTS

2.0 LIMITING CONDITIONS FOR OPERATION

2.1 Reactor Coolant System (Continued)

2.1.1 Operable Components (Continued)

(c) For the purposes of items (a) and (b) above, the containment spray pumps can be considered as available shutdown cooling pumps only if both of the following conditions are met:

(i) Reactor Coolant System temperature is less than 120°F.

(ii) The Reactor Coolant System is vented with a vent area equal to or greater than 47 in^2.

Exceptions

All decay heat removal loops may be made inoperable for up to 8 hours provided (1) no operations are permitted that would cause dilution of the reactor coolant system boron concentration, (2) no refueling operations are taking place, and (3) all containment penetrations providing direct access from the containment atmosphere to the outside atmosphere are closed within 4 hours.

(5) At least one reactor coolant pump or one low pressure safety injection pump in the shutdown cooling mode shall be in operation whenever a change is being made in the boron concentration of the reactor coolant when fuel is in the reactor.

(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 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 unless at least one of the following conditions is met:

(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.

In the hot shutdown mode, a single reactor coolant loop provides sufficient heat removal capability for removing decay heat; however, single failure considerations require that two loops be operable.

In the cold shutdown mode, 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.

The requirement that at least one shutdown cooling loop be in operation during refueling ensures that: (1) sufficient cooling capacity is available to remove decay heat and maintain the water in the reactor pressure vessel below 210°F as required during the refueling mode, and (2) sufficient coolant circulation is maintained through the reactor core to minimize the effects of a boron dilution incident and prevent boron stratification.

The requirement to have two shutdown cooling pumps operable when there is less than 15 feet of water above the core ensures that a single failure of the operating shutdown cooling loop will not result in a complete loss of decay heat removal capability. With the reactor vessel head removed and 15 feet of water above the core, a large heat sink is available for core cooling; thus, in the event of a failure of the operating shutdown cooling loop, adequate time is provided to initiate emergency procedures to cool 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 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.

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.

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

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.

The exception to Specification 2.1.1(4) requiring all containment penetrations providing direct access, from the containment to the outside atmosphere be closed within 4 hours requires that the equipment hatch be closed and held in place by a minimum of four bolts.

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.

Objective

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

Specification

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

(1) 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 Figure 2-1A.

(2) 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 Figures 2-1B.

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

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

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

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

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

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

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

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

(a) The curve in Figure 2-3 shall be used to predict the increase in transition temperature based on integrated fast neutron flux. If measurements on the irradiation specimens indicate a deviation from this curve, a new curve shall be constructed.

(b) The limit line on the figures shall be updated for a new integrated power period as follows: the total integrated reactor thermal power from startup to the end of the new period shall be converted to an equivalent integrated fast neutron exposure ($E \geq 1$ MeV). 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.

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.

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

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

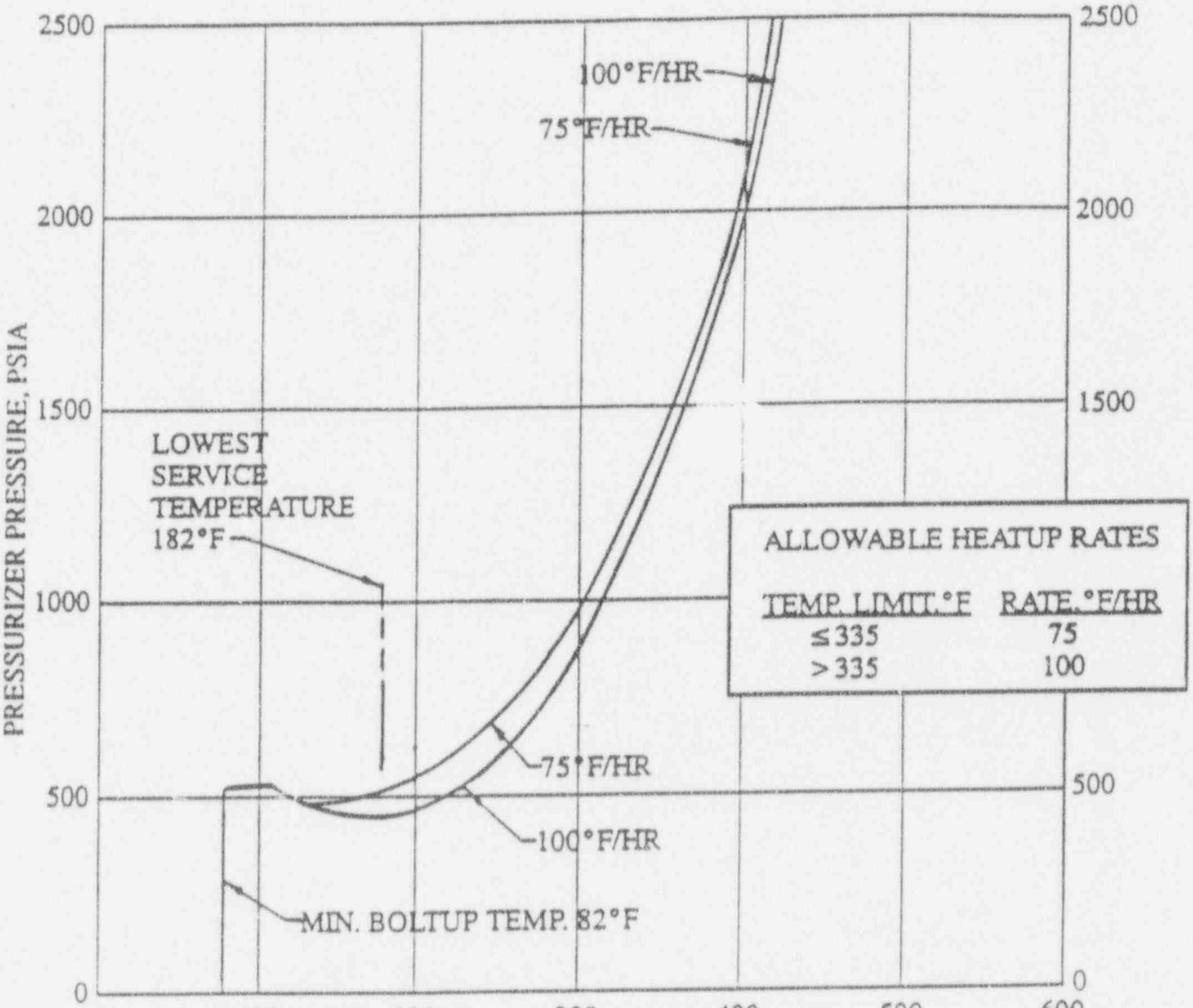
K_{IM} = Stress intensity factor for membrane stress (pressure). The 2 represents a safety factor of 2 on pressure.

K_{IT} = Stress intensity factor radial thermal gradient.

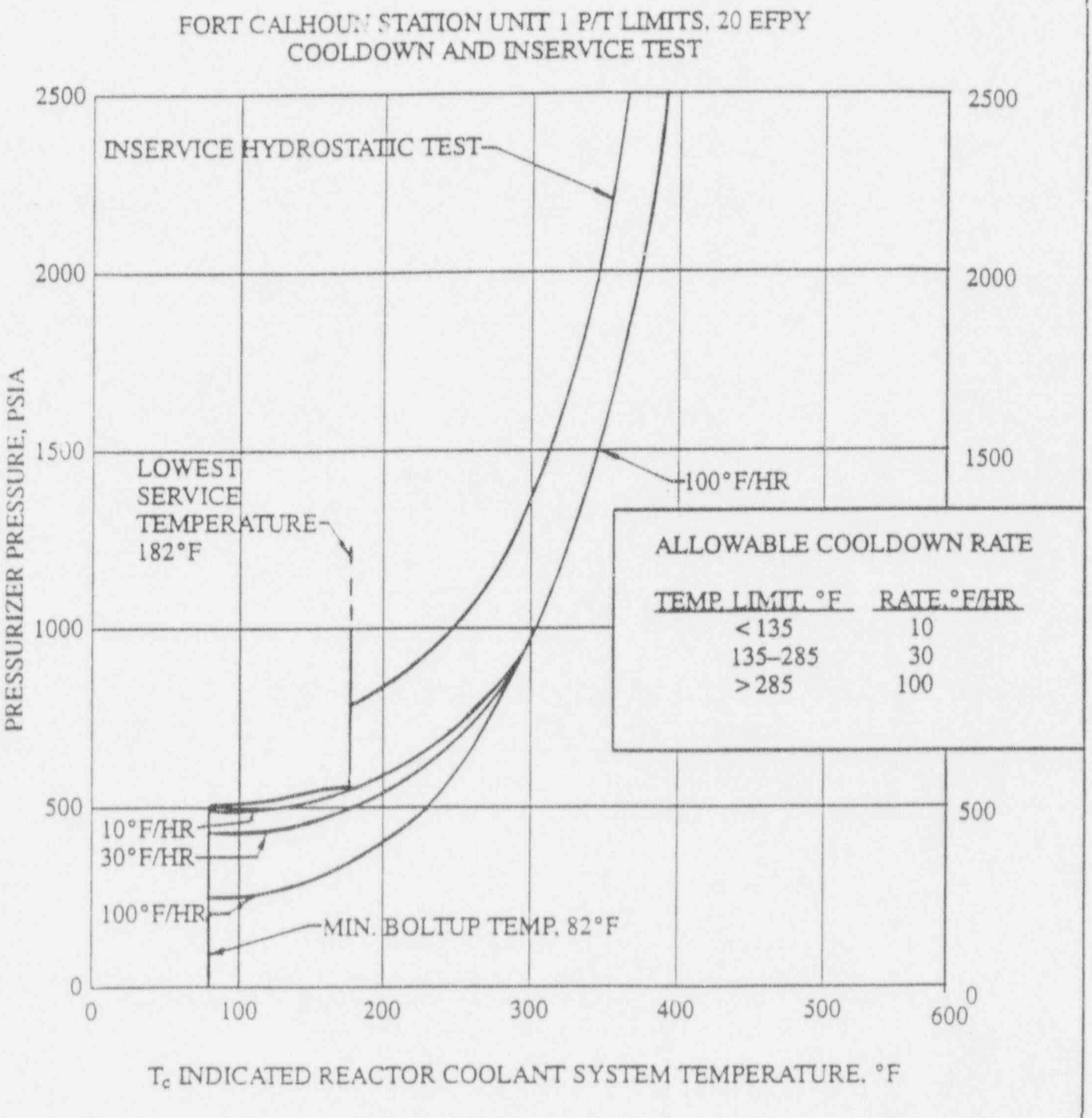
The above equation is applied to the reactor vessel beltline. For plant heatup the reference stress intensity in calculated for both the 1/4t and 3/4t locations. Composite curves are then generated for each heatup rate by combining the most restrictive pressure-temperature limits over the complete temperature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

For plant cooldown thermal and pressure stress are additive.

FORT CALHOUN STATION UNIT 1 P/T LIMITS, 20 EFPY



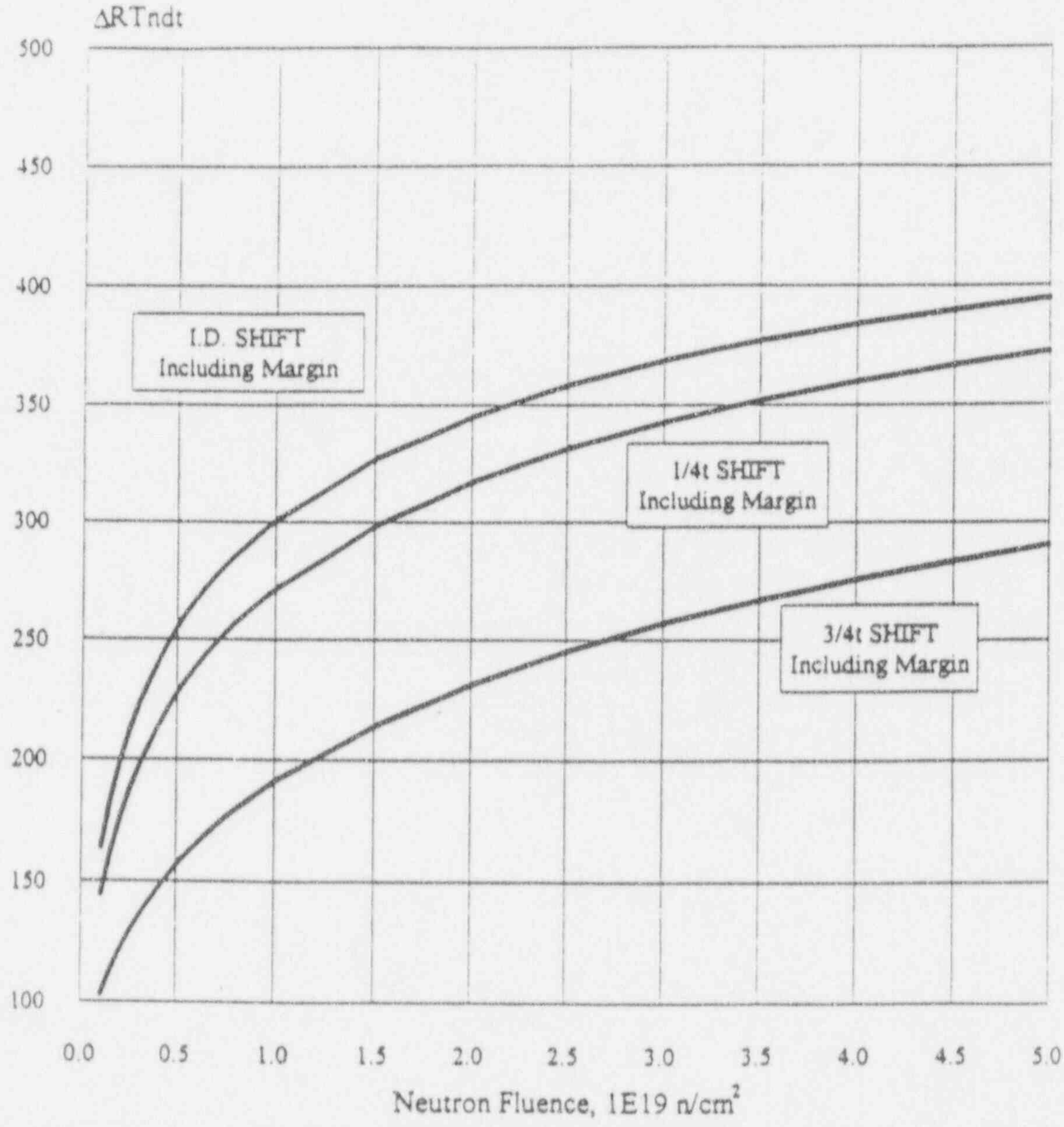
RCS Pressure-Temperature Limits for Heatup	Omaha Public Power District Fort Calhoun Station- Unit No. 1	Figure 2-1A



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

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

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

$$K_{IR} = AP + B$$

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

K_{IR} is then varied as a function of temperature from Figure G-2110-1 of ASME III and the allowable pressure calculated. 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.

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 2500 psia $\pm$1% and 2545 psia $\pm$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) Whenever the reactor is in power operation, eight of the ten main steam safety valves shall be operable with their lift settings adjusted to ensure valves on each header opening at 1000 psia +3/-2%, 1015 psia +3/-2%, 1025 psia +3/-2%, 1040 psia +3/-2%, and 1050 psia +3/-2%.[1]

(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 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 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.

d. With both PORVs inoperable in Modes 4 or 5 depressurize and vent the RCS through at least a 0.94 square inch or larger vent within the next 36 hours.

(5) Two power-operated relief valves (PORVs) and their associated block valves shall be operable in Modes 1, 2, and 3.

a. With one or both PORV(s) inoperable because of excessive seat leakage, within 1 hour either restore the PORV(s) to operable status or close the associated block valve(s) with power maintained to the block valve(s); otherwise, be in at least HOT SHUTDOWN within the next 6 hours and in COLD SHUTDOWN within the following 36 hours.

b. With one PORV inoperable due to causes other than excessive seat leakage, within 1 hour either restore PORV to operable status or close its associated block valve and remove power from the block valve; restore the PORV to operable status within the following 72 hours or be in HOT SHUTDOWN within the next 6 hours and in COLD SHUTDOWN with the following 36 hours.

c. With both PORVs inoperable due to causes other than excessive seat leakage, within 1 hour either restore at least one PORV to operable status or close both block valves, remove power from the block valves, and be in HOT SHUTDOWN within the next 6 hours and in COLD SHUTDOWN within the following 36 hours.

d. With one or both block valve(s) inoperable, within 1 hour restore the block valve(s) to operable status or place the associated PORV(s) in the closed position. Restore at least one block valve to operable status within the next hour if both block valves are inoperable; restore the remaining inoperable block valve to operable within 72 hours. Otherwise, be in at least HOT SHUTDOWN within the next 6 hours and in COLD SHUTDOWN within the following 36 hours.

Basis

The highest reactor coolant system pressure reached in any of the accidents analyzed resulted from a complete loss of turbine generator load without simultaneous reactor trip while operating at 1500 MWt.[2] This pressure was less than the 2750 psia safety limit and the ASME Section III upset pressure limit of 10% greater than the design pressure.[1] The reactor is assumed to trip on a "High Pressurizer Pressure" trip signal.

The pressurizer safety valves are required to be calibrated to within $\pm$ 1% of the specified setpoint value using ASME Section XI test methods. ASME Section XI requires that valves in steam service use steam as the test medium for establishing the setpoint. With the presence of a water-filled loop seal, establishing the valve setpoint with steam may result in in-situ valve actuation at pressures outside the $\pm$ 1% tolerance specified. Under transient conditions, it is expected that the valve(s) will actuate at no less than 4% below, nor greater than 6% above, the specified setpoint, which is within the tolerance assumed in the safety analysis.[2]

The power-operated relief valves (PORVs) operate to relieve RCS pressure below the setting of the pressurizer code safety valves. These relief valves have remotely operated block valves to provide a positive shutoff capability should a relief valve become inoperable. The electrical power for both the relief valves and the block valves is capable of being supplied from an emergency power source to ensure the ability to seal this possible RCS leakage path.

Action statements (5)b. and c. include the removal of power from a closed block valve to preclude any inadvertent opening of the block valve at a time the PORV may not be closed due to maintenance. However, the applicability requirements of the LCO to operate with the block valve(s) closed with power maintained to the block valve(s) are only intended to permit operation of the plant for a limited period of time not to exceed the next refueling shutdown (Mode 5), so that maintenance can be performed on the PORV(s) to eliminate the seat leakage condition.

To determine the maximum steam flow, the only other pressure relieving system assumed operational is the main steam safety valves. Conservative values for all systems parameters, delay times and core moderator coefficients are assumed. Overpressure protection is provided to portions of the reactor coolant system which are at the highest pressure considering pump head, flow pressure drops and elevation heads.

If no residual heat were removed by any of the means available, the amount of steam which could be generated at safety valve lift pressure would be less than half of the capacity of one safety valve. This specification, therefore, provides adequate defense against overpressurization when the reactor is subcritical.

Performance of certain calibration and maintenance procedures on safety valves requires removal from the pressurizer. Should a safety valve be removed, either operability of the other safety valve or maintenance of at least one nozzle open to atmosphere will assure that sufficient relief capacity is available. Use of plastic or other similar material to prevent the entry of foreign material into the open nozzle will not be construed to violate the "open to atmosphere" provision, since the presence of this material would not significantly restrict the discharge of reactor coolant.

The total relief capacity of the ten main steam safety valves is 6.54×10^6 lb/hr. If, following testing, the as found setpoints are outside +/-1% of nominal nameplate values, the valves are set to within the +/-1% tolerance. The main steam safety valves were analyzed for a total loss of main feedwater flow while operating at 1500 MWt[3] to ensure that the peak secondary pressure was less than 1100 psia, the ASME Section III upset pressure limit of 10% greater than the design pressure. At the power of 1500 MWt, sufficient relief valve capacity is available to prevent overpressurization of the steam system on loss-of-load conditions.[4]

The power-operated relief valve low setpoint will be adjusted to provide sufficient margin, when used in conjunction with Technical Specification Sections 2.1.1 and 2.3, to prevent the design basis pressure transients from causing an overpressurization incident. Limitation of this requirement to scheduled cooldown ensures that, should emergency conditions dictate rapid cooldown of the reactor coolant system, inoperability of the low temperature overpressure protection system would not prove to be an inhibiting factor. The effective full flow area of an open PORV is 0.94 in^2.

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

References

(1) Article 9 of the 1968 ASME Boiler and Pressure Vessel Code, Section III
(2) USAR, Section 14.9
(3) USAR, Section 14.10
(4) USAR, Sections 4.3.4, 4.3.9.5

(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 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.

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.

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

The limits for the safety injection tank pressure and volume assure the required amount of water injection during an accident and are based on values used for the accident analyses. The minimum 116.2 inch level corresponds to a volume of 825 ft^3 and the maximum 128.1 inch level corresponds to a volume of 895.5 ft^3. Prior to the time the reactor is brought critical, the valving of the safety injection system must be checked for correct alignment and appropriate valves locked. Since the system is used for shutdown cooling, the valving will be changed and must be properly aligned prior to start-up of the reactor.

be available for emergency core cooling, but the contents of one of the tanks is assumed to be lost through the reactor coolant system. In addition, of the three high-pressure safety injection pumps and the two low-pressure safety injection pumps, for large break analysis it is assumed that two high pressure and one low pressure operate while only one of each type is assumed to operate in the small break analysis[(5)]; and also that 25% of their combined discharge rate is lost from the reactor coolant system out of the break. The transient hot spot fuel clad temperatures for the break sizes considered are shown in USAR Section 14.

The restriction on HPSI pump operability at low temperatures, in combination with the PORV setpoints ensure that the reactor vessel pressure-temperature limits would not be exceeded in the case of an inadvertent actuation of the operable HPSI and charging pumps.

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

Technical Specification 2.2(1) specifies that, when fuel is in the reactor, at least one flow path shall be provided for boric acid injection to the core. Should boric acid injection become necessary, and no charging pumps are operable, operation of a single HPSI pump would provide the required flow path. The HPSI pump flow rate must be restricted to that of three charging pumps in order to minimize the consequences of a mass addition transient while at low temperatures.

MINIMUM FREQUENCIES FOR CHECKS, CALIBRATIONS AND TESTING OF MISCELLANEOUS INSTRUMENTATION AND CONTROLS

	Channel Description	Surveillance Function	Frequency	Surveillance Method
19.	Auxiliary Feedwater Flow	Check	M	Channel check.
		Calibrate	R	Known pressure inputs.
20.	Subcooled Margin Monitor	Check	M	Channel check.
		Calibrate	R	Known pressure inputs and known resistance substituted for RTD inputs.
21.	PORV Operation and Acoustic Position Indication	Check	M	Channel check.
		Calibrate	R	Apply acoustic input.
		Verify	R	Operation on emergency power supply.
22.	PORV Block Valve Operation and Position Indication	Check	Q	Cycle valve. Valve is exempt from testing when it has been closed to comply with LCO Action Statement 2.1.6(5)a.
		Calibrate	R	Check valve stroke against limit switch position.
		Verify	R	Operability on emergency power supply.
23.	Safety Valve Acoustic Position Indication	Check	M	Circuit check.
		Calibration	R	Apply acoustic input.
24.	PORV/Safety Valve Tail Pipe Temperature	Check	M	Circuit check.
		Calibrate	R	Apply known input.