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

OMAHA PUBLIC POWER DISTRICT

DOCKET NO. 50-285

FORT CALHOUN STATION, UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 47
License No. DPR-40

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

   A. The application for amendment by Omaha Public Power District (the licensee) filed July 17, 1979, as supplemented by letters dated October 30 and December 4, 1979, and January 28, February 12, 25, and March 12, 1980, 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 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, the license 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:

   Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 47, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of the date of its issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

W. P. Gammill

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Director for Operating Reactor Projects
Division of Operating Reactors

Attachment:
Changes to the Technical
Specifications

Date of Issuance: April 1, 1980

ATTACHMENT TO LICENSE AMENDMENT NO. 47

FACILITY OPERATING LICENSE NO. DPR-40

DOCKET NO. 50-285

Replace the following pages and figures of the Appendix "A" Technical Specifications with the enclosed pages and figures. The revised pages are identified by Amendment number and contain vertical lines indicating the area of change.

Remove Pages	Insert Pages
1-1	1-1
1-2	1-2
1-8	1-8
1-10	1-10
2-1	2-1
2-4 thru 2-7	2-4 thru 2-7
2-7a	2-7a
2-15 and 2-16	2-15 and 2-16
2-22	2-22
2-23	2-23
2-23a	2-23a and b
2-50	2-50
2-50f	2-50f
2-54	2-54
2-55	2-55
2-55A thru 2-55E	2-55A thru 2-55E
2-56	2-56
2-57	2-57
2-57a and b	2-57a and b
2-57f	2-57f

Remove Figures	Insert Figures
1-1 thru 1-7	1-1 thru 1-7
2-1A and B	2-1A and B
2-2A and B	2-2A and B
2-3	2-3
2-5 thru 2-9	2-5 thru 2-9

1.0 SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

1.1 Safety Limits - Reactor Core

Applicability

This specification applies to the limiting combinations of reactor power and reactor coolant system flow, temperature and pressure during operation.

Objective

To maintain the integrity of the fuel cladding and prevent the release of significant amounts of fission products to the reactor coolant.

Specifications

The reactor power level shall not exceed the allowable limit for the pressurizer pressure and the cold leg temperatures as shown in Figure 1-1 for 4-pump operation. The safety limit is exceeded if the point defined by the combination of reactor coolant cold leg temperature and power level is at any time above the appropriate pressurizer pressure line.

Basis

To maintain the integrity of the fuel cladding and prevent the release of significant amounts of fission products to the reactor coolant, it is necessary to prevent overheating of the cladding under normal operating conditions. This is accomplished by operating within the nucleate boiling regime of heat transfer, wherein the heat transfer coefficient is large enough so that the clad surface temperature is only slightly greater than the coolant saturation temperature. The upper boundary of the nucleate boiling regime is termed "departure from nucleate boiling" (DNB).

At this point there is a sharp reduction of the heat transfer coefficient, which would result in high clad temperature and the possibility of clad failure. Although DNB is not an observable parameter during reactor operation, the observable parameters of reactor thermal power and reactor coolant flow, temperature and pressure can be related to DNB through the W-3 DNB correlation.(1) The W-3 DNB correlation has been developed to predict DNB and the location of DNB for axially uniform and non-uniform heat flux distributions. The local DNB ratio (DNBR), defined as the ratio of the heat flux that

1.0 SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

1.1 Safety Limits - Reactor Core (Continued)

would cause DNB at a particular core location to the actual heat flux at that location, is indicative of the margin to DNB. The minimum value of the DNBR during steady state operation, normal operational transients, and anticipated transients is limited to 1.30. A DNBR of 1.30 corresponds to a 95% probability at a 95% confidence level that DNB will not occur, which is considered an appropriate margin to DNB for all operating conditions.(1)

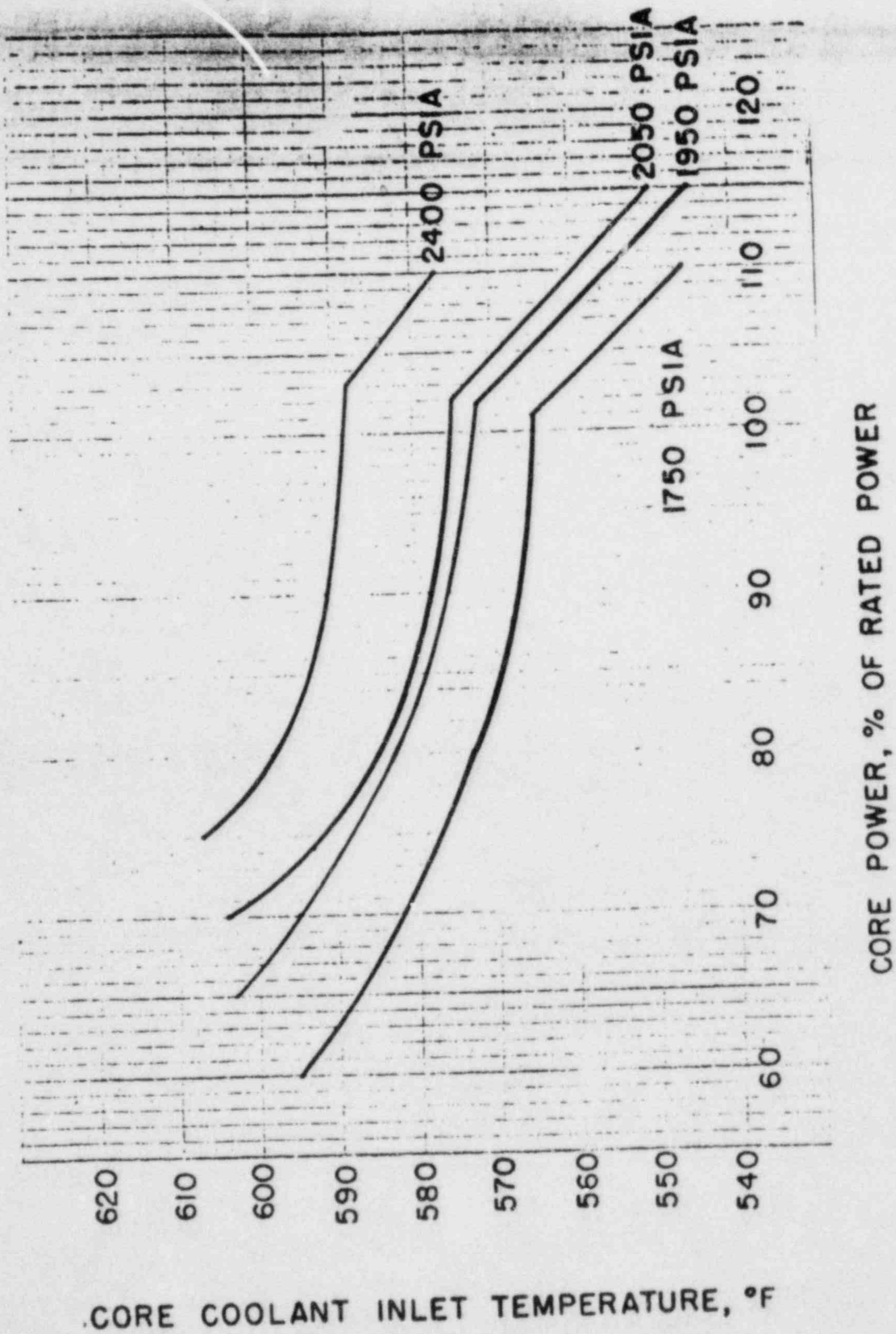
The curves of Figure 1-1 represent the loci of points of reactor thermal power (either neutron flux instruments or ΔT instruments), reactor coolant system pressure, and cold leg temperature for which the DNBR is 1.30. The area of safe operation is below these lines.

The reactor core safety limits are based on radial peaks limited by the CEA insertion limits in Section 2-10 and axial shapes within the axial power distribution trip limits in Figure 1-2 and a total unrodded planar radial peak of 1.62. The LSSS in Figure 1-3 is based on the assumption that the unrodded integrated total radial peak (F_R^T) is 1.57. This peaking factor is slightly higher (more conservative) than the maximum predicted unrodded total radial peak during core life, excluding measurement uncertainty.

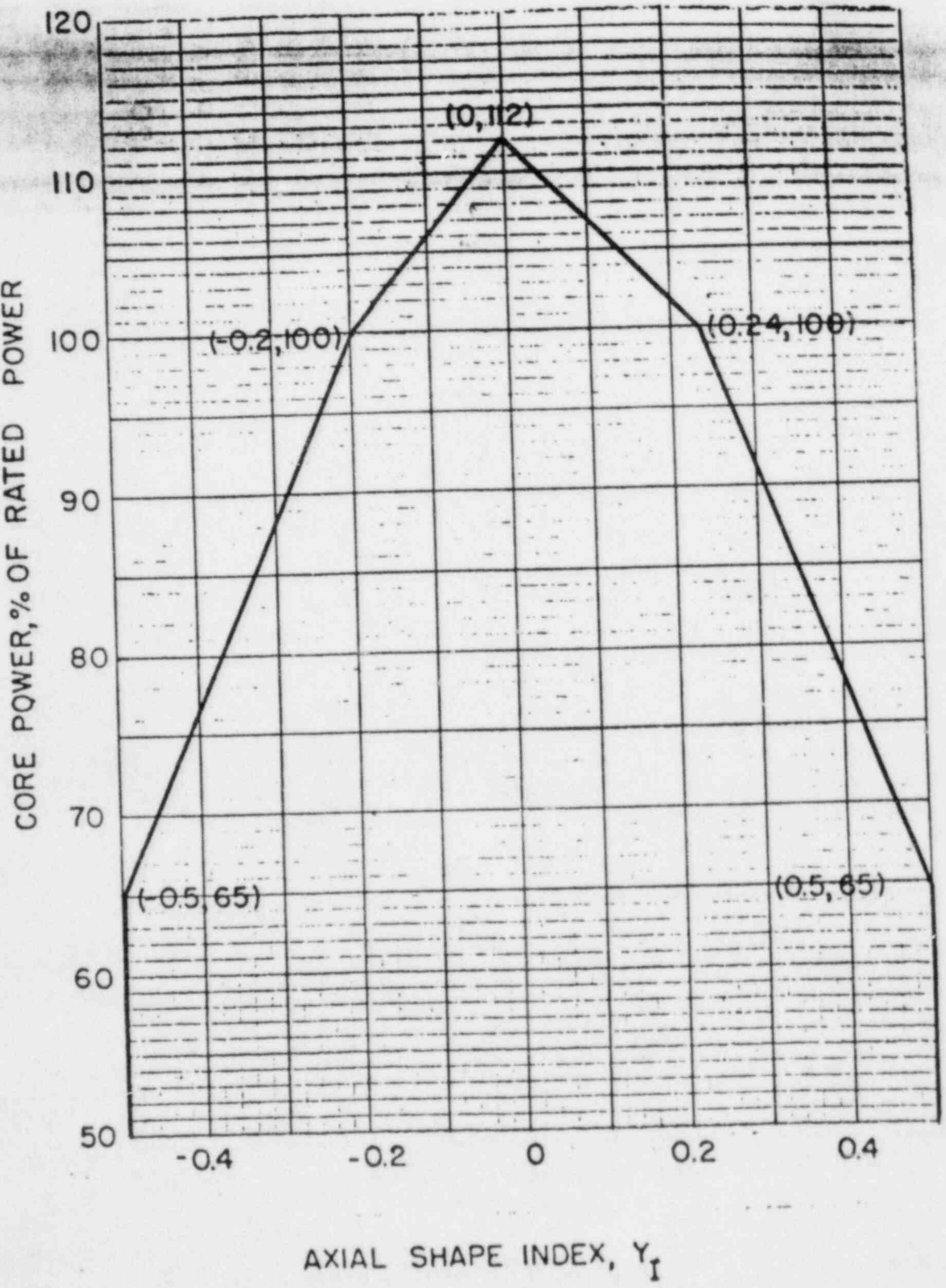
Flow maldistribution effects for operation under less than full reactor coolant flow have been evaluated via model tests.(3) The flow model data established the maldistribution factors and hot channel inlet temperature for the thermal analyses that were used to establish the safe operating envelopes presented in Figure 1-1. The reactor protective system is designed to prevent any anticipated combination of transient conditions for reactor coolant system temperature, pressure and thermal power level that would result in a DNBR of less than 1.30.(4)

References

(1) FSAR, Section 3.5.5
(2) FSAR, Section 3.5.2
(3) FSAR, Section 1.4.6
(4) FSAR, Section 3.5.3



FORT CALHOUN TECHNICAL SPECIFICATIONS	THERMAL MARGIN/LOW PRESSURE SAFETY LIMITS, 4 PUMP OPERATION	FIGURE 1-1



FORT CALHOUN TECHNICAL SPECIFICATIONS	O.P.P.D. FORT CALHOUN AXIAL POWER DISTRIBUTION L.S.S.S. FOR 4 PUMP OPERATION	FIGURE 1-2

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FORT CALHOUN TECHNICAL SPECIFICATIONS

THERMAL MARGIN/LOW PRESSURE LSSS, 4 PUMP OPERATION

FIGURE 1-3

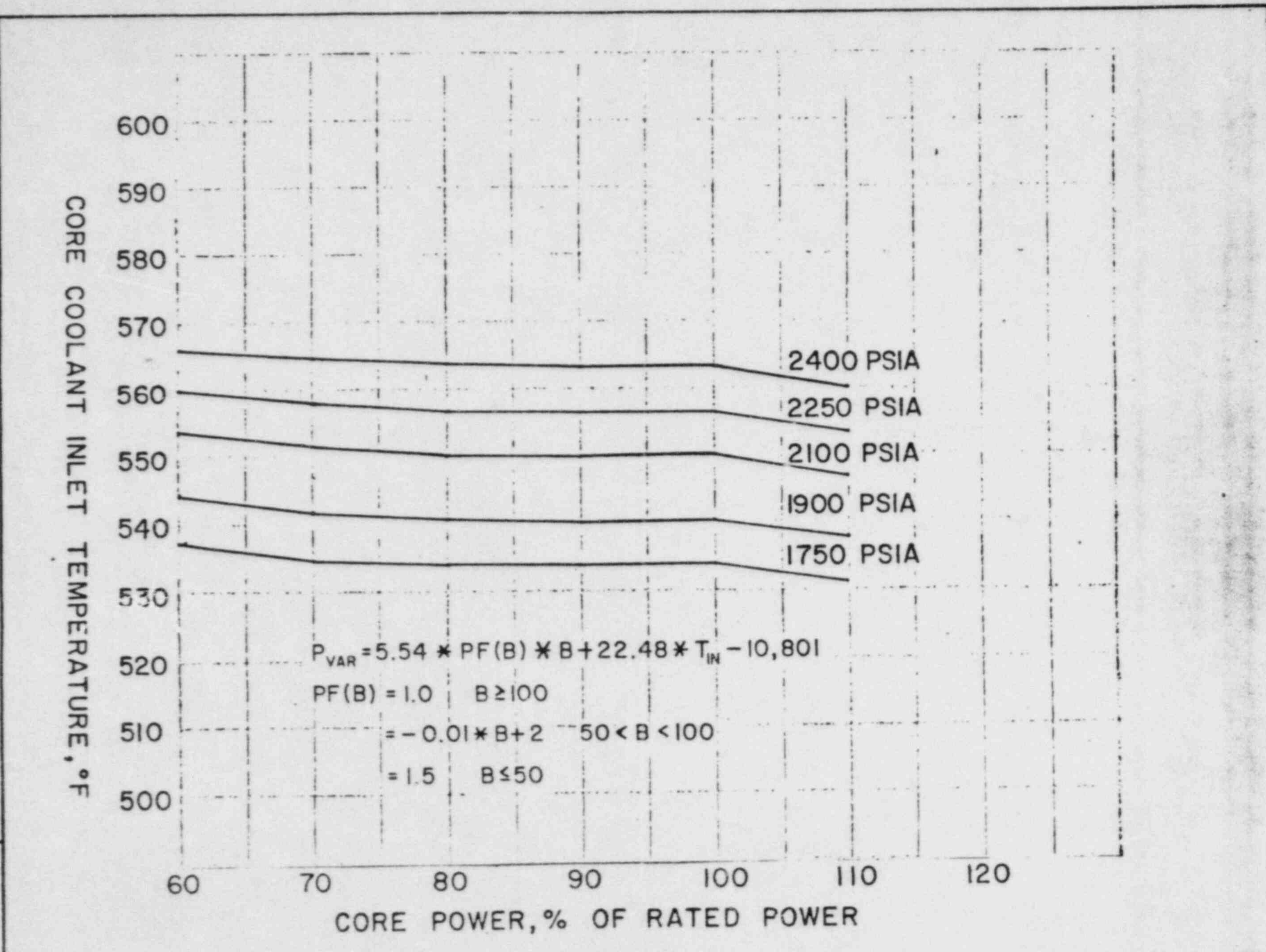

FIGURE 1-4

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FIGURE 1-5

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FIGURE 1-6

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FIGURE 1-7

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(3) High Pressurizer Pressure - A reactor trip for high pressurizer pressure is provided in conjunction with the reactor and steam system safety valves to prevent reactor coolant system overpressure (Specification 2.1.6). In the event of loss of load without reactor trip, the temperature and pressure of the reactor coolant system would increase due to the reduction in the heat removed from the coolant via the steam generators. The power-operated relief valves are set to operate concurrently with the high pressurizer pressure reactor trip. This setting is 100 psi below the nominal safety valve setting (2500 psia) to avoid unnecessary operation of the safety valves. This setting is consistent with the trip point assumed in the accident analysis.[1]

(4) Thermal Margin/Low Pressure Trip - The thermal margin/low pressure trip is provided to prevent operation when the DNBR is less than 1.3, including allowance for measurement error. The thermal and hydraulic limits shown on Figure 1-3 define the limiting values of reactor coolant pressure, reactor inlet temperature, and reactor power level which ensure that the thermal criteria (8) are not exceeded. The low set point of 1750 psia trips the reactor in the unlikely event of a loss-of-coolant accident. The thermal margin/low pressure trip set points shall be set according to the formula given on Figure 1-3. The variables in the formula are defined as:

B = High auctioneered thermal (ΔT) or nuclear power in % of rated power.
T_{IN} = Core inlet temperature, $^{\circ}F$.
P_{VAR} = Reactor pressure, psia.

TABLE 1.1

RPS LIMITING SAFETY SYSTEM SETTINGS

No.	Reactor Trip	Trip Setpoints
1	High Power Level (A)	
	4-Pump Operation	≤107.0% of Rated Power
	3-Pump Operation	≤45% of Rated Power
	2-Pump Operation	≤30% of Rated Power
2	Low Reactor Coolant Flow (B) (F)	
	4-Pump Operation	≥95% of 4 Pump Flow
	3-Pump Operation	≥71% of 4 Pump Flow
	2-Pump Operation	≥46% of 4 Pump Flow
3	Low Steam Generator Water Level	31.2% of Scale (Top of feedwater ring; 4'10" below normal water level)
4	Low Steam Generator Pressure (C)	≥500 psia
5	High Pressurizer Pressure	≤2400 psia
6	Thermal Margin/Low Pressure (B) (F)	1750 psia to 2400 psia (depending on the reactor coolant temperature as shown in Figure 1-3)
7	High Containment Pressure (D)	≤5 psig
8	Axial Power Distribution (E)	(Figure 1-2)

2.0 LIMITING CONDITIONS FOR OPERATION

2.1 Reactor Coolant System

2.1.1 Operable Components

Applicability

Applies to the operable status of the reactor coolant system components.

Objective

To specify certain conditions of the reactor coolant system components which must be met to assure safe reactor operation.

Specifications

Limiting conditions for operation are as follows:

(1) 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.

(2) Except for special tests during initial start-up testing, at least two reactor coolant pumps shall be in operation whenever the reactor is critical.

(3) Minimum flow conditions for various reactor power levels shall be maintained as shown in Table 1-1.

(4) 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 in-service inspection program specified in Section 3.3(2) prior to exceeding a reactor coolant temperature of 300°F.

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

(6) Reactor coolant system leak and hydrostatic tests shall be conducted within the limitations of Figures 2-1A and 2-1B.

(7) Maximum secondary hydrostatic test pressure shall not exceed 1250 psia. A minimum temperature of 82°F is required. Only 10 cycles are permitted.

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

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

(b) The limit line on the figures shall be updated for a new integrated power period as follows: the total integrated reactor thermal power from startup to the end of the new period shall be converted to an equivalent integrated fast neutron exposure ($E \geq 1$ MeV). For this plant, based upon surveillance materials tests, the predicted surface fluence at the reactor vessel belt-line weld material for 40 years at 1500 MWt and an 80% load factor is 4.4×10^{19} n/cm^2. The predicted transition temperature shift to the end of the new period shall then be obtained from Figure 2-3.

(c) The limit lines in Figure 2-1A through 2-2B shall be moved parallel to the temperature axis (horizontal) in the direction of increasing temperature a distance equivalent to the transition temperature shift during the period since the curves were last constructed. The boltup temperature limit line shall remain at 82°F as it is set by the NDTT of the reactor vessel flange and not subject to fast neutron flux. The lowest service temperature shall remain at 162°F because components related to this temperature are also not subject to fast neutron flux.

Basis

All components in the reactor coolant system are designed to withstand the effects of cyclic loads due to reactor coolant system temperature and pressure changes.(1) These cyclic loads are introduced by normal unit load transients, reactor trips and startup and shutdown operation.

During unit startup and shutdown, the rates of temperature and pressure changes are limited. The design number of cycles for heatup and cooldown is based upon a rate of 100°F in any one hour period and for cyclic operation.

The maximum allowable reactor coolant system pressure at any temperature is based upon the stress limitations for brittle fracture considerations. These limitations are derived by using the rules contained in Section III(2) of the ASME Code including Appendix G, Protection Against Nonductile Failure and the rules contained in 10CFR50, 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.

Similar testing was not performed on all remaining material in the reactor coolant system. However, sufficient impact testing was performed to meet appropriate design code requirements(3) and a conservative RT_{NDT} of 50°F has been established.

As a result of fast neutron irradiation in the region of the core, there will be an increase in the T_{NDT} with operation. The techniques used to predict the integrated fast neutron ($E \geq 1$ MeV) fluxes of the reactor vessel are described in Section 3.4.6 of the FSAR, except that the integrated fast neutron flux ($E \geq 1$ MeV) is 4.4×10^{19} n/cm^2, including tolerance, over the 40 year design life of the vessel.(5)

Since the neutron spectra and the flux measured at the samples and reactor vessel inside radius should be nearly identical, the measured transition shift for a sample can be applied to the adjacent section of the reactor vessel for later stages in plant life equivalent to the difference in calculated flux magnitude. The maximum exposure of the reactor vessel will be obtained from the measured sample exposure by application of the calibrated azimuthal neutron flux variation. The maximum integrated fast neutron ($E \geq 1$ MeV) exposure of the reactor vessel including tolerance is computed to be 4.4×10^{19} n/cm^2 for 40 years operation at 1500 MWt and 80% load factor.(5) The predicted T_{NDT} shift for an integrated fast neutron ($E \geq 1$ MeV) exposure of 4.4×10^{19} n/cm^2 is 350°F, the value obtained from the curve shown in Figure 2-3. The actual shift in T_{NDT} will be re-established periodically during 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 FSAR. To compensate for any increase in the T_{NDT} caused by irradiation, limits on the pressure-temperature relationship are periodically changed to stay within the stress limits during heatup and cooldown. Analysis of the first removed irradiated reactor vessel surveillance specimen has shown that the fluence at the end of fuel Cycle 6 will be 8.94×10^{18} n/cm^2 on the inside surface of the reactor vessel.(5) This corresponds to 5.2 Effective Full Power Years (EFPY) at 1500 MWt and a total shift of the RT_{NDT} of 223°F as determined from Figure 2-3.

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

The limit lines in Figure 2-1A through 2-2B are based on the following:

A. Heatup and Cooldown Curves - From Section III of the ASME Code Appendix G-2215.

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

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

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

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

The above equation is applied to the reactor vessel beltline. For plant heatup the thermal stress is opposite in sign from the pressure stress and consideration of a heatup rate would allow for a higher pressure. For heatup it is therefore conservative to consider an isothermal heatup or $K_{IT} = 0$.

For plant cooldown thermal and pressure stress are additive.

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

M_M = ASME III, Figure G-2214-1

P = Pressure, psia

R = Vessel Radius - in.

t = Vessel Wall Thickness - in.

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

M_T = ASME III, Figure G-2214-2

ΔT_W = Highest Radial Temperature Gradient Through Wall at End of Cooldown

K_{IT} is therefore calculated at a maximum gradient and is considered a constant = A for cooldown and zero for heatup. $\frac{M_M}{t}$ R is also a constant = B.

Therefore:

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

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

K_{IR} is then varied as a function of temperature from Figure G-2110-1 of ASME III and the allowable pressure calculated. Hydrostatic head (48 psi) and instrumentation errors (12°F and 32 psi) are considered when plotting the curves.

B. System Hydrostatic Test - The system hydrostatic test curve is developed in the same manner as in A above with the exception that a safety factor of 1.5 is allowed by ASME III in lieu of 2.

C. Lowest Service Temperature = 50°F + 100°F + 12°F = 162°F. As indicated previously, an RT_{NDT} for all material with the exception of the reactor vessel beltline was established at 50°F. ASME III, Art. NB-2332(b) requires a lowest service temperature of RT_{NDT} + 100°F for piping, pumps and valves. Below this temperature a pressure of 20 percent of the system hydrostatic test pressure (.20)(3125) - 48 - 32 psi = 545 psia cannot be exceeded.

D. Boltup Temperature = 10°F + 60°F + 12°F = 82°F. At pressure below 545 psia, a minimum vessel temperature must be maintained to comply with the manufacturer's specifications for tensioning the vessel head. This temperature is based on previous NDTT methods. This temperature corresponds to the measured 10°F NDTT of the reactor vessel flange, which is not subject to radiation damage, plus 60°F data scatter in NDTT measurements, plus 12°F instrument error.

E. Reactor Critical Heatup and Cooldown Figures. During low power physics testing, the reactor may be made critical at reduced temperature and pressure. To provide for heatup and cooldown during testing, Appendix G requires that the RCS temperature be increased an additional 40°F beyond heatup and cooldown curves for the non-critical reactor. Also, Appendix G requires that the RCS temperature must be greater than the minimum temperature, 363°F, required for the 3125 psia hydrostatic testing to 125% of the 2500 psia RCS design pressure.

References

(1) FSAR, Section 4.2.2

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

(3) FSAR, Section 4.2.4

(4) FSAR, Section 3.4.6

(5) Omaha Public Power District, Fort Calhoun Station Unit No. 1, Evaluation of Irradiated Capsule W-225, May, 1979.

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2.1.6 Pressurizer and Steam System Safety Valves

Applicability

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

Objective

To specify minimum requirements pertaining to the pressurizer and steam system 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 between 2500 psia and 2545 psia ±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 steam safety valves shall be operable with their lift settings between 1000 psia and 1050 psia with a tolerance of ±1% of the nominal nameplate set point values.(1)

(4) Both pressurizer power-operated relief valves (PORV's) shall be operable during scheduled heatup and cooldown to prevent violation of the pressure-temperature limits designated by Figures 2-1A and 2-1B. One PORV may be inoperable for up to 7 days, provided the remaining PORV is operable. If the above conditions of this paragraph cannot be met, the primary system must be depressurized and vented.

Basis

The highest reactor coolant system pressure reached in any of the accidents analyzed was 2480 psia and resulted from a complete loss of turbine generator load without simultaneous reactor trip while operating at 1500 MWt.(2) The reactor is assumed to trip on a "High Pressurizer Pressure" trip signal.

To determine the maximum steam flow, the only other pressure relieving system assumed operational is the steam system 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 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 steam system safety valves is 6.54×10^6 lb/hr. At the power of 1500 MWt, sufficient relief valve capacity is available to prevent overpressurization of the steam system on loss-of-load conditions.

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.

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) FSAR, Section 14.9

(3) FSAR, 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 reactor vessel head, a pressurizer safety valve, or a PORV is removed.

Whenever the reactor coolant system cold leg temperature is below 320°F, at least one (1) HPSI pump shall be disabled.

Whenever the reactor coolant system cold leg temperature is below 310°F, at least two (2) HPSI pumps shall be disabled.

Whenever the reactor coolant system cold leg temperature is below 276°F, all three (3) HPSI pumps shall be disabled.

In the event that no charging pumps are operable, a single HPSI pump may be made operable and utilized for boric acid injection to the core.

Basis

The normal procedure for starting the reactor is to first heat the reactor coolant to near operating temperature by running the reactor coolant pumps. The reactor is then made critical by withdrawing CEA's and diluting boron in the reactor coolant. With this mode of start-up, the energy stored in the reactor coolant during the approach to criticality is substantially equal to that during power operation and therefore all engineered safety features and auxiliary cooling systems are required to be fully operable. During low power physics tests at low temperatures, there is a negligible amount of stored energy in the reactor coolant; therefore, an accident comparable in severity to the design basis accident is not possible and the engineered safeguards systems are not required.

The SIRW tank contains a minimum of 283,000 gallons of usable water containing at least 1700 ppm boron.(1) This is sufficient boron concentration to provide a shutdown margin of 5%, including allowances for uncertainties, with all control rods withdrawn and a new core at a temperature of 60°F.(2)

The limits for the safety injection tank pressure and volume assure the required amount of water injection during an accident and are based on values used for the accident analyses. The minimum 116.2 inch level corresponds to a volume of 825 ft^3 and the maximum 128.1 inch level corresponds to a volume of 895.5 ft^3.

Prior to the time the reactor is brought critical, the valving of the safety injection system must be checked for correct alignment and appropriate valves locked. Since the system is

2.0 LIMITING CONDITIONS FOR OPERATION

2.3 Emergency Core Cooling System (Continued)

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

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

If a component is found to be inoperable, it will be possible in most cases to effect repairs and restore the system to full operability within a relatively short time. For a single component to be inoperable does not negate the ability of the system to perform its function, but it reduces the redundancy provided in the reactor design and thereby limits the ability to tolerate additional equipment failures. To provide maximum assurance that the redundant component(s) will operate if required to do so, the redundant component(s) is to be tested prior to initiating repair of the inoperable component. If it develops that the inoperable component is not repaired within the specified allowable time period, or a second component in the same or related system is found to be inoperable, the reactor will initially be put in the hot shutdown condition to provide for reduction of cooling requirements after a postulated loss-of-coolant accident. This will also permit improved access for repairs in some cases. After a limited time in hot shutdown, if the malfunction(s) is not corrected, the reactor will be placed in the cold shutdown condition utilizing normal shutdown and cooldown procedures. If the cold shutdown condition, release of fission products or damage of the fuel elements is not considered possible.

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

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

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

2.0 LIMITING CONDITIONS FOR OPERATION

2.3 Emergency Core Cooling System (Continued)

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

The LOCA analysis confirms adequate core cooling for the break spectrum up to and including the 32 inch double-ended break assuming the safety injection capability which most adversely affects accident consequences and are defined as follows. The entire contents of all four safety injection tanks are assumed to be available for emergency core cooling, but the contents of one of the tanks is assumed to be lost through the reactor coolant system. In addition, of the three high-pressure safety injection pumps and the two low-pressure safety injection pumps, for large break analysis it is assumed that two high pressure and one low pressure operate while only one of each type is assumed to operate in the small break analysis(5); and also that 25% of their combined discharge rate is lost from the reactor coolant system out of the break. The transient hot spot fuel clad temperatures for the break sizes considered are shown on FSAR Figures 1-19 (Amendment No. 34).

Inadvertent actuation of three (3) HPSI pumps and three (3) charging pumps, coincident with the opening of one of the two PORV's, would result in a peak primary system pressure of 1190 psia. 1190 psia corresponds with a minimum permissible temperature of 320°F on Figure 2-1B. Thus, at least one HPSI pump is disabled at 320°F.

Inadvertent actuation of two (2) HPSI pumps and three (3) charging pumps, coincident with the opening of one of the two PORV's, would result in a peak primary system pressure of 1040 psia. 1040 psia corresponds with a minimum permissible temperature of 310°F on Figure 2-1B. Thus, at least two HPSI pumps will be disabled at 310°F.

Inadvertent actuation of one (1) HPSI and three (3) charging pumps, coincident with opening of one of the two PORV's, would result in a peak primary system pressure of 685 psia. 685 psia corresponds with a minimum allowable temperature of 276°F on Figure 2-1B. Thus, all three HPSI pumps will be disabled at 276°F.

Inadvertent actuation of three (3) charging pumps, coincident with opening of one of the two PORV's, would result in a peak primary system pressure of 160 psia. 160 psia corresponds with a minimum allowable temperature of 78°F (approximately the boltup temperature of 82°F) on Figure 2-1B. Thus, disabling of the charging pumps is not required.

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.

2.0 LIMITING CONDITIONS FOR OPERATION

2.3 Emergency Core Cooling System (Continued)

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.

References

(1) FSAR, Section 14.15.1

(2) FSAR, Section 6.2.3.1

(3) FSAR, Section 14.15.3

(4) FSAR, Appendix K

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

2.10.2 Reactivity Control Systems and Core Physics Parameters Limits

Applicability

Applies to operation of control element assemblies and monitoring of selected core parameters whenever the reactor is in cold or hot shutdown, hot standby, or power operation conditions.

Objective

To ensure (1) adequate shutdown margin following a reactor trip, (2) the MTC is within the limits of the safety analysis, and (3) control element assembly operation is within the limits of the setpoint and safety analysis.

Specification

(1) Shutdown Margin with T_{cold} >210°F

Whenever the reactor is in hot shutdown, hot standby or power operation conditions, the shutdown margin shall be $\geq$3.0% Δk/k. With the shutdown margin <3.0% Δk/k, initiate and continue boration until the required shutdown margin is achieved.

(2) Shutdown Margin with T_{cold} $\leq$210°F

Whenever the reactor is in cold shutdown conditions, the shutdown margin shall be $\geq$2.0% Δk/k. With the shutdown margin <2.0% Δk/k, initiate and continue boration until the required shutdown margin is achieved.

(3) Moderator Temperature Coefficient

The moderator temperature coefficient (MTC) shall be:

a. Less positive than +0.2 x 10^{-4} Δρ/°F including uncertainties for power levels at or above 80% of rated power.

b. Less positive than +0.5 x 10^{-4} Δρ/°F including uncertainties for power levels below 80% of rated power.

c. More positive than -2.3 x 10^{-4} Δρ/°F including uncertainties at rated power.

With the moderator temperature coefficient confirmed outside any one of the above limits, change reactivity control parameters to bring the extrapolated MTC value within the above limits within 3 hours or be in at least hot shutdown within 6 hours.

1. The total available shutdown margin may be reduced to 2% Δk/k during the measurement of the shutdown CEA group reactivities, or

2. The total available shutdown margin may be reduced to the worth of the worst stuck CEA's during the measurement of the stuck CEA reactivity.

(ii) If the shutdown margin specified in part (i) above is not available immediately, initiate and continue boration until the requirements of 2.10.2(1) are met.

(iii) The shutdown margin specified in part (i) above shall be verified every 8 hour shift.

c. Moderator Temperature Coefficient

(i) The moderator temperature coefficient (MTC) requirements of 2.10.2(3) may be suspended during physics tests at less than 10^{-1}% of rated power.

(ii) If power exceeds 10^{-1}% of rated power, either:

1. Reduce power to less than 10^{-1}% of rated power within 15 minutes, or

2. Be in hot shutdown in 2 hours.

Basis

Shutdown Margin

A sufficient shutdown margin ensures that (1) the reactor can be made subcritical from all operating conditions, (2) the reactivity transients associated with postulated accident conditions are controllable within acceptable limits, and (3) the reactor will be maintained sufficiently subcritical to preclude inadvertent criticality in the shutdown condition.

Shutdown margin requirements vary throughout core life as a function of fuel depletion, RCS boron concentration, and RCS T_{avg}. The most restrictive condition occurs at EOL, with T_{avg} at no load operating temperature, and is associated with a postulated steam line break accident and resulting uncontrolled RCS cooldown. In the analysis of this accident, a minimum shutdown margin of 3.0% Δk/k is initially adequate to control the reactivity transient. Accordingly, 2.10.3 In Core Instrumentation

Applies to the operability and alarm values of the rhodium detector in-core instruments system.

Objective

To specify the functional requirements on the use of the rhodium in-core instrumentation system for (1) the recalibration of the ex-core detector inputs to the axial power distribution trip calculators and (2) monitoring of kw/ft and power distribution.

Specification

(1) A minimum of four in-core locations at each detector level (or a total of 16 detectors) with at least one location in the center seven rows of fuel assemblies and at least one location outside the center seven rows of fuel assemblies shall be operable during recalibration of the ex-core detectors.

(2) The in-core detector system shall be operable with:

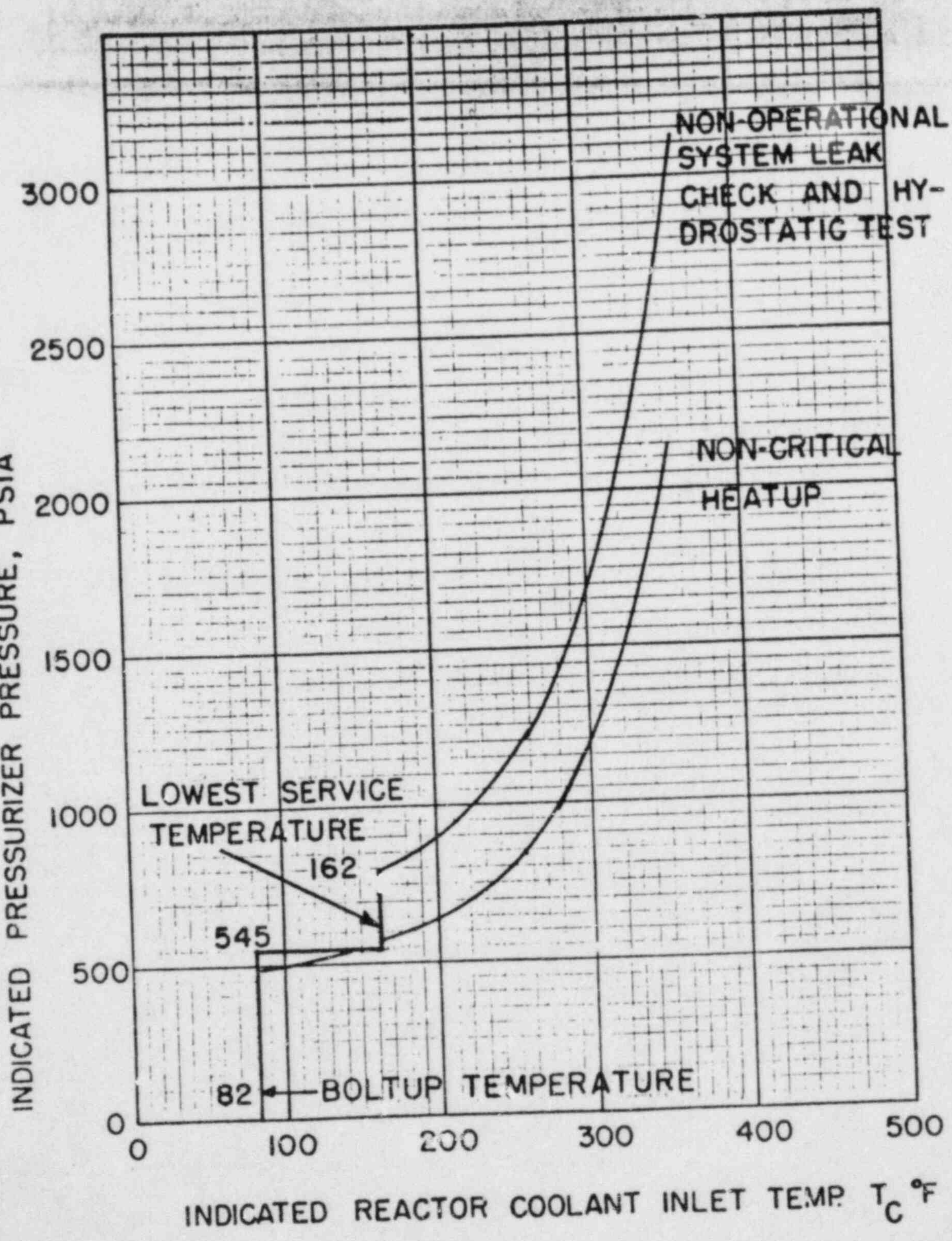
(a) At least 75% of all in-core detector strings, and

(b) A minimum of two in-core detector strings per full axial length quadrant

whenever the in-core system is used to monitor F_{xy}^T, F_R^T, the radial power distribution and the peak linear heat rate. An operable in-core detector string shall consist of three or more operable rhodium detectors. With the in-core detector system inoperable, do not use the system for the above applicable monitoring functions.

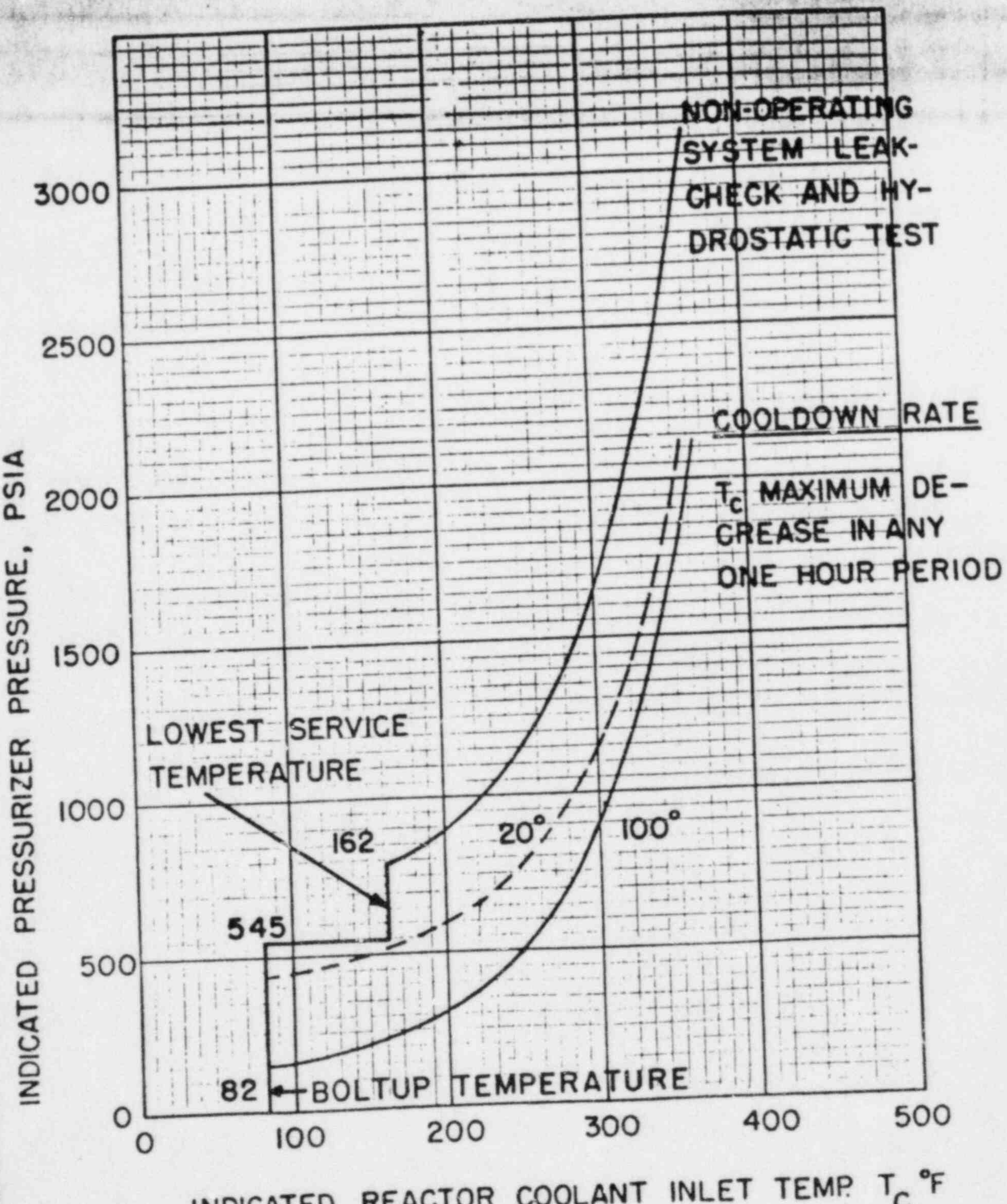
(3) If the recalibration of the ex-core detectors has not been accomplished within the previous 30 equivalent full power days, reduce the axial power distribution monitoring limits and trip setpoints, Figures 2-6, 2-7, and 1-2, by 0.03 ASI units. If the recalibration of the ex-core detectors has not been accomplished within the previous 200 equivalent full power days, the power shall be limited to less than that corresponding to 75% of the peak linear heat rate permitted by Specification 2.10.4.(1).

(4) After each fuel loading, the incore detector system shall be operable with at least 75% of the incore detector strings operable and a minimum of two quadrant symmetric incore detector string locations per core quadrant for the initial measurement of the linear heat rate, F_R^T, F_{xy}^T and the azimuthal power tilt.



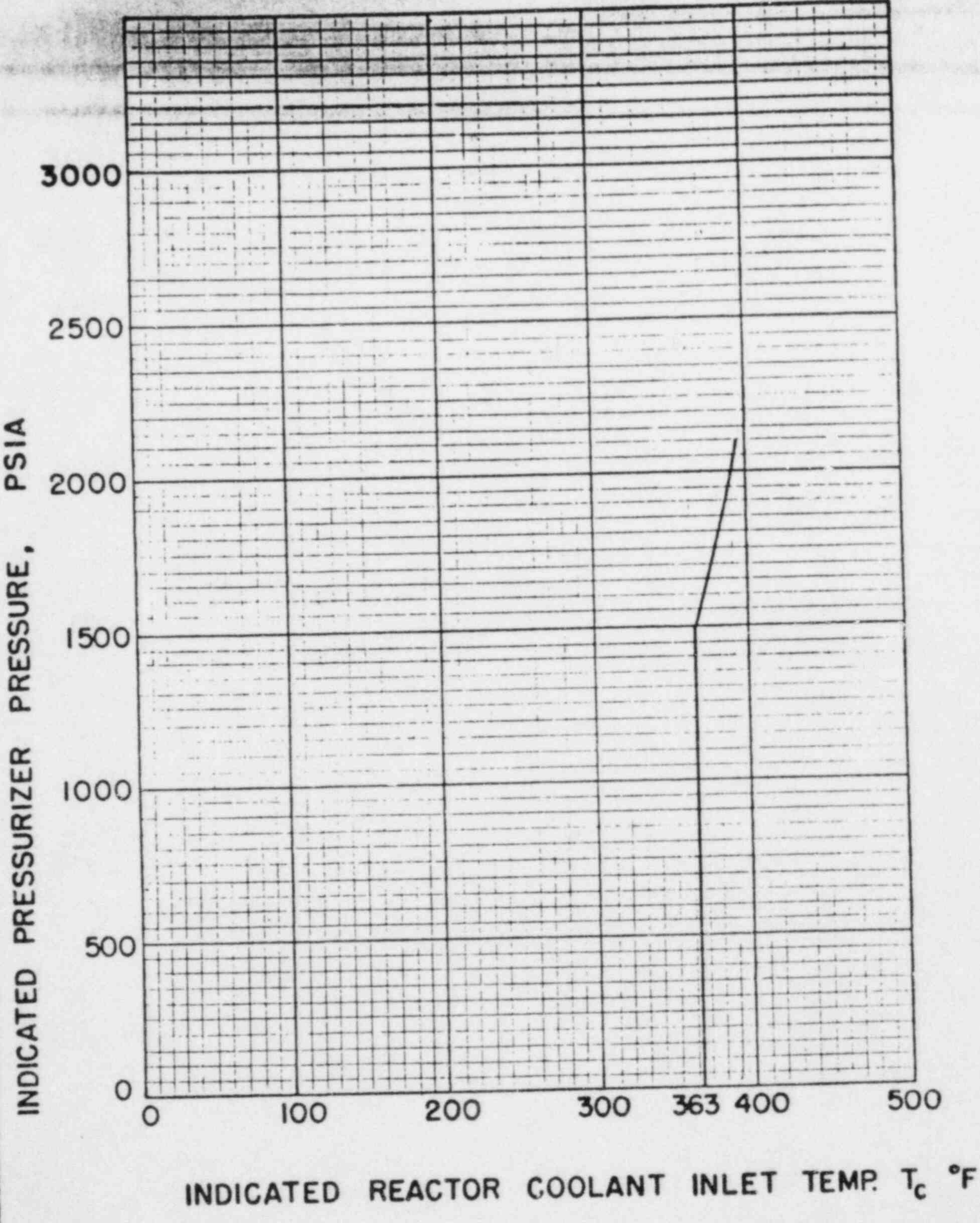
FORT CALHOUN TECHNICAL SPECIFICATIONS	REACTOR COOLANT SYSTEM PRESSURE-TEMPERATURE LIMITATIONS HEATUP 4.3 TO 5.2 EFPY OPER.-REACTOR NOT CRIT.	FIGURE 2-1A

Amendment No. ~~77~~, 47



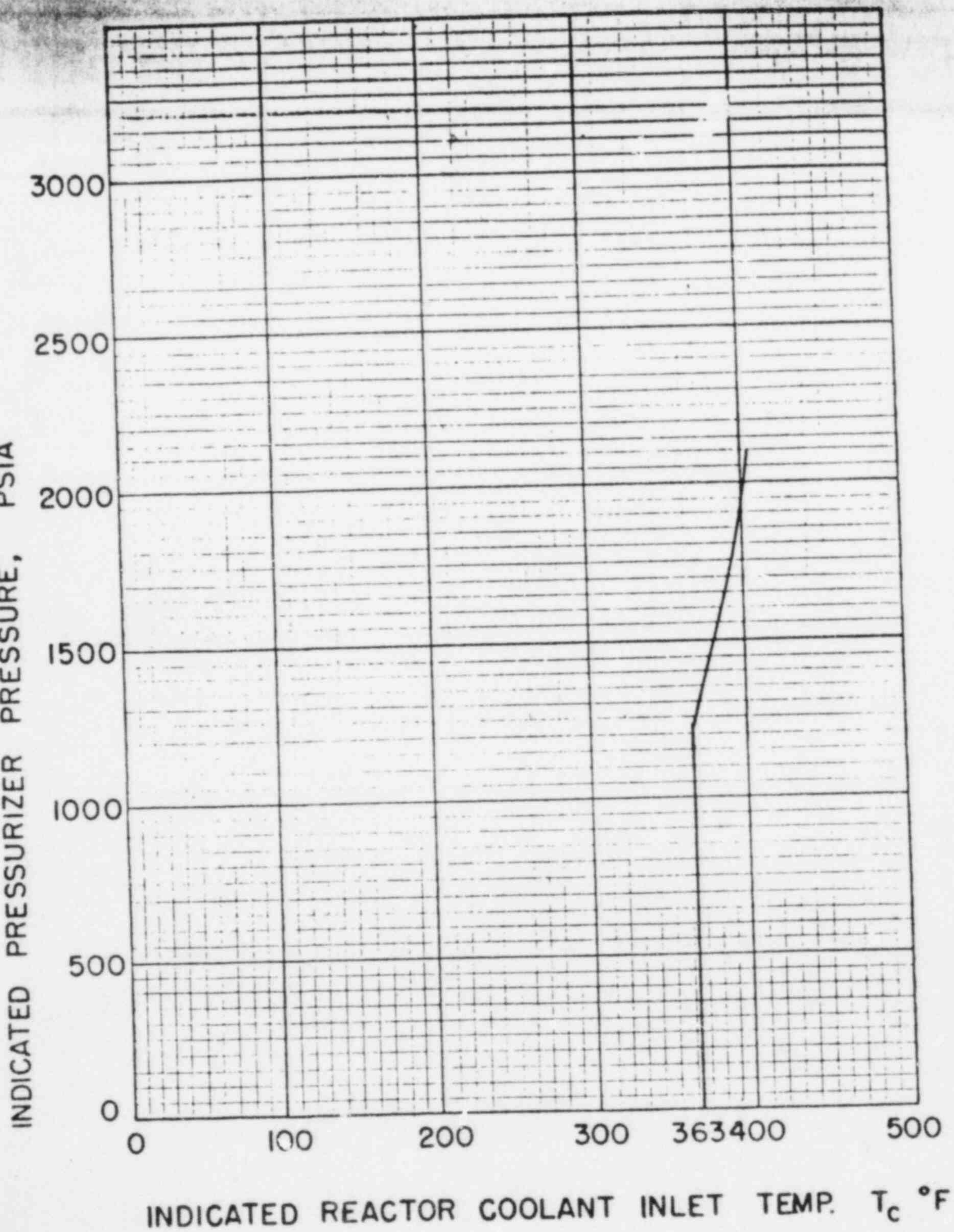
FORT CALHOUN TECHNICAL SPECIFICATIONS	REACTOR COOLANT SYSTEM PRESSURE-TEMPERATURE LIMITATIONS COOLDOWN 4.3 TO 5.2 EFPY OPER.-REACTOR NOT CRIT.	FIGURE 2-1B

. Amendment No. 22, 47



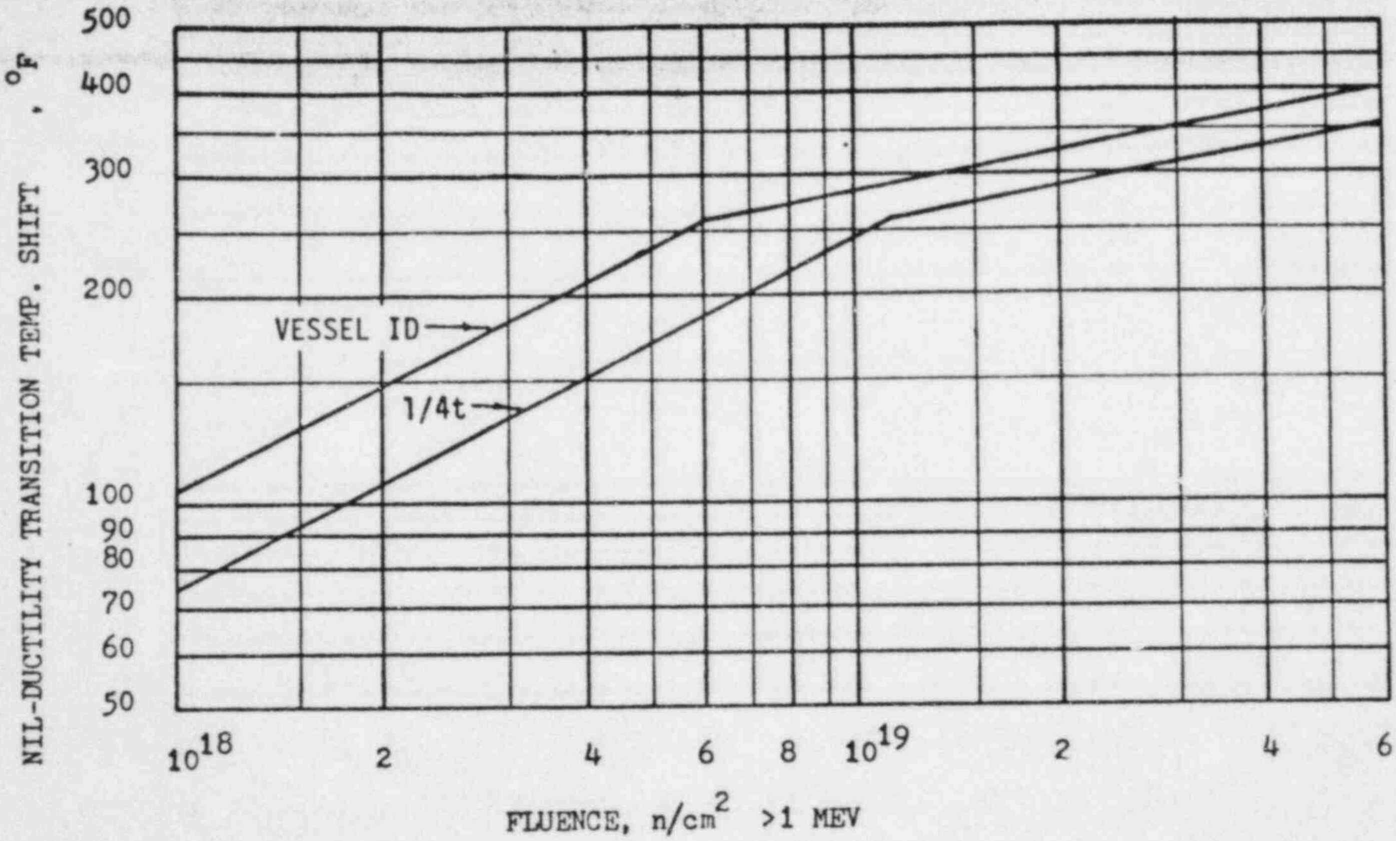
FORT CALHOUN TECHNICAL SPECIFICATIONS	REACTOR COOLANT SYSTEM PRESSURE - TEMPERATURE LIMITATIONS HEATUP 4.3 TO 5.2 EFPY OPER.- REACTOR CRITICAL	FIGURE 2-2A

Amendment No. 22, 47



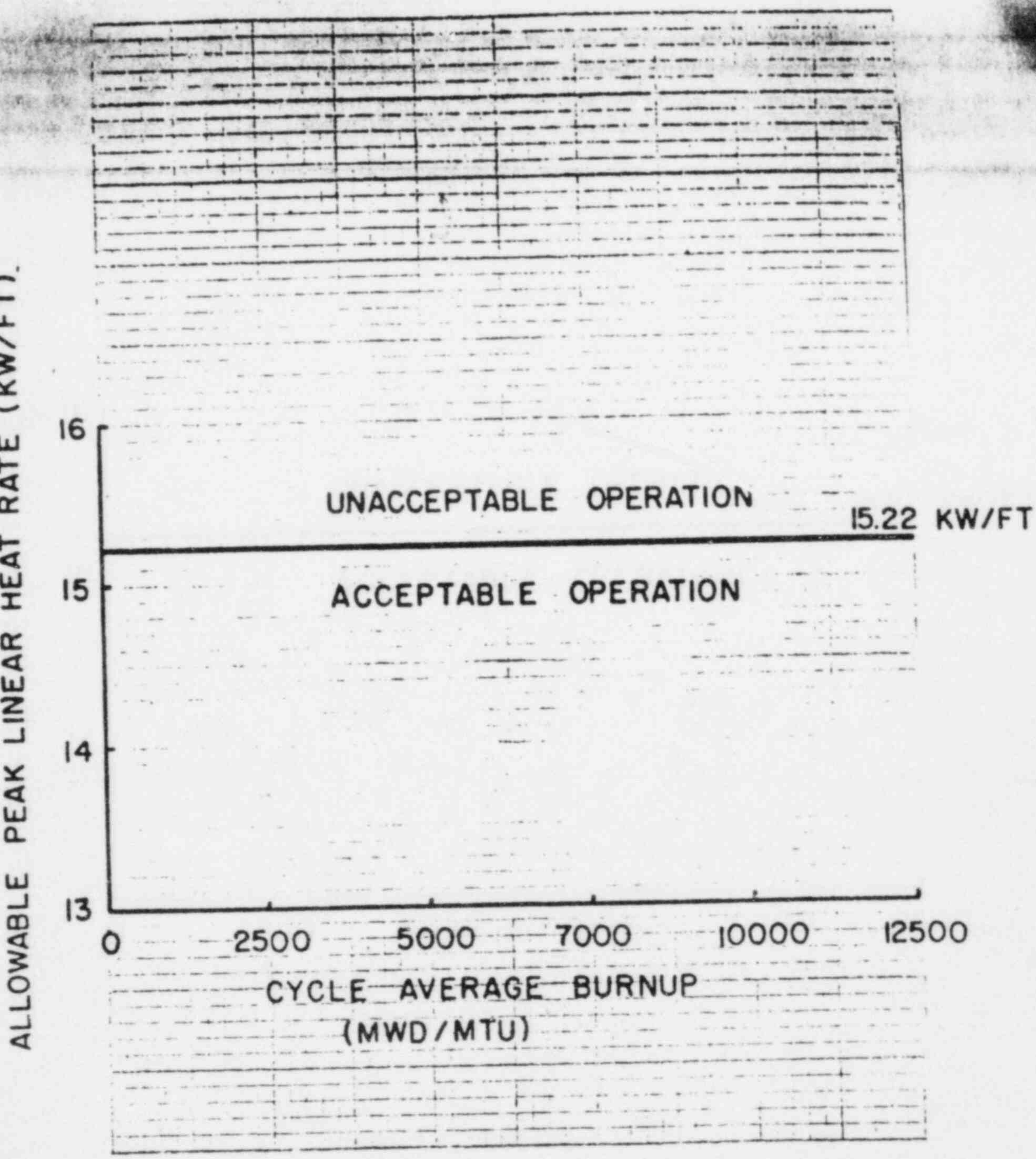
FORT CALHOUN TECHNICAL SPECIFICATIONS	REACTOR COOLANT SYSTEM PRESSURE — TEMPERATURE LIMITATIONS COOLDOWN 4.3 TO 5.2 EFPY OPER.—REACTOR CRITICAL	FIGURE 2-2B

Amendment No. 22,47



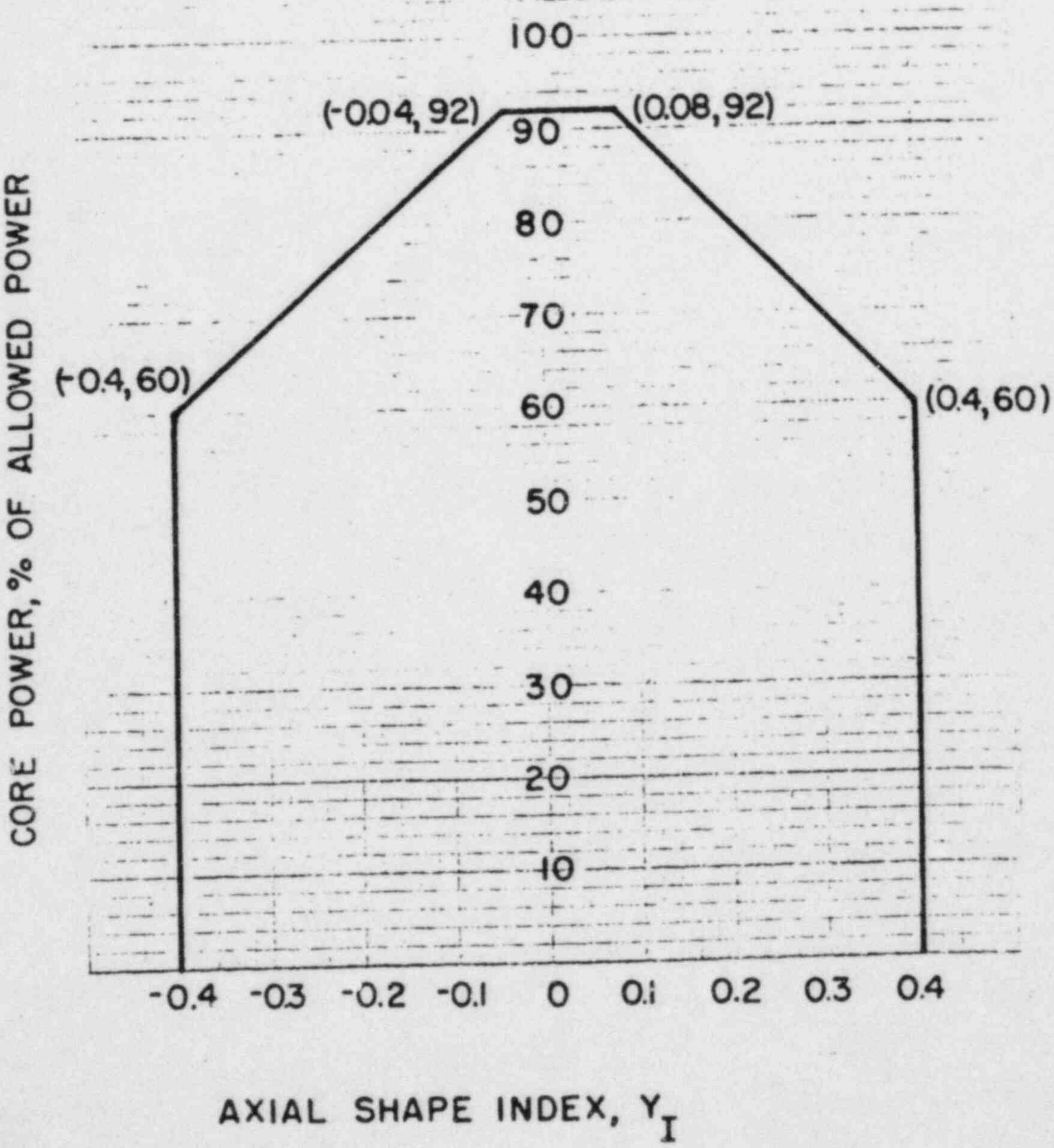
THE PREDICTED NDTT SHIFT OF THE QUARTER THICKNESS CURVE ($\frac{1}{4}$t) IS USED TO DETERMINE THE CORRESPONDING TEMPERATURE SHIFT OF FIGURES 2-1A, 2-1B, 2-2A, AND 2-2B.

FORT CALHOUN TECHNICAL SPECIFICATION	PREDICTED NDTT SHIFT FOR THE REACTOR VESSEL BELTLINE, DERIVED FROM REG. GUIDE 1.99-1	FIGURE 2-3



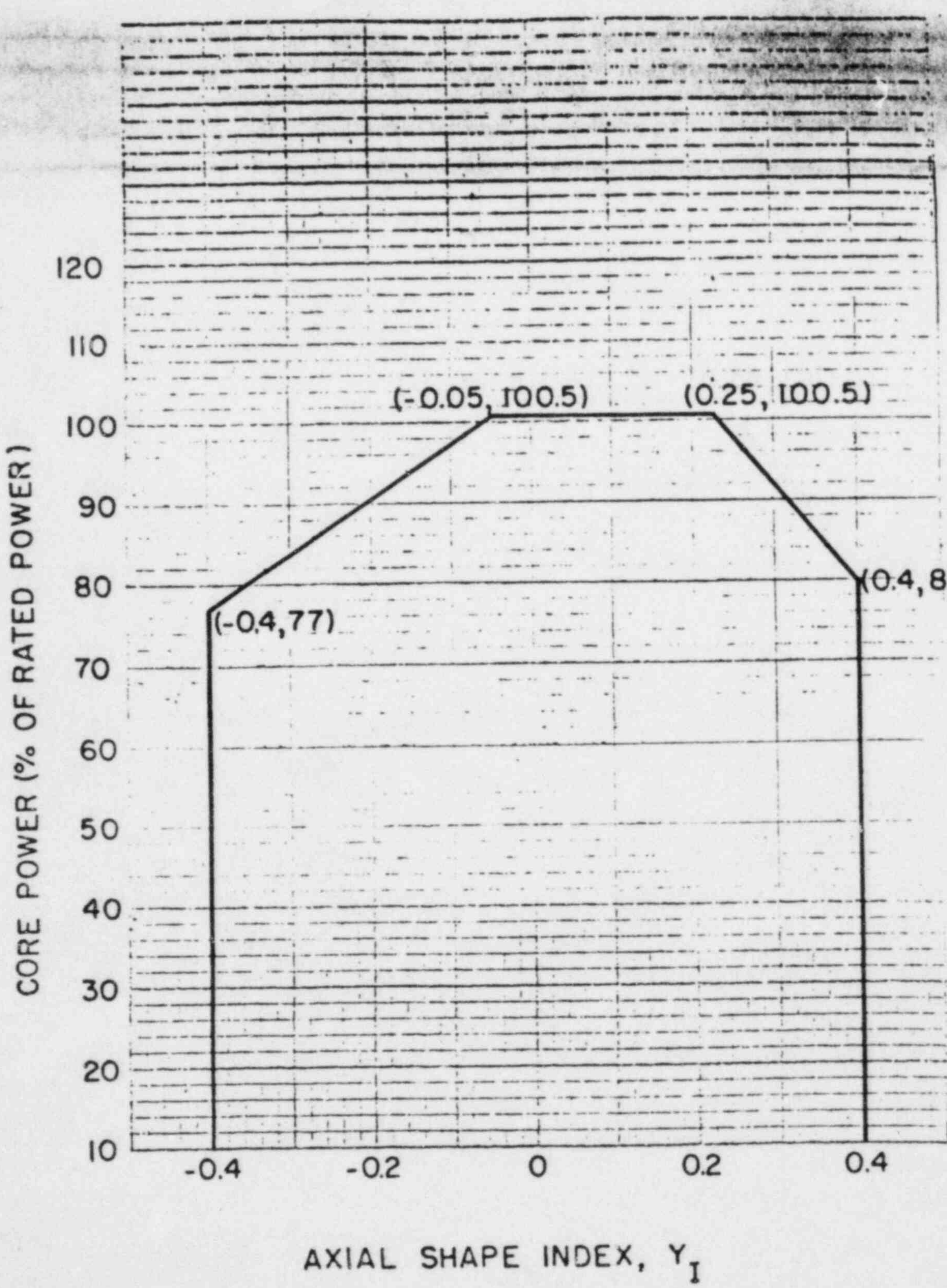
FORT CALHOUN TECHNICAL SPECIFICATIONS	ALLOWABLE PEAK LINEAR HEAT RATE VS. BURNUP	FIGURE 2-5

Amendment No. 8, 20, 32, 47



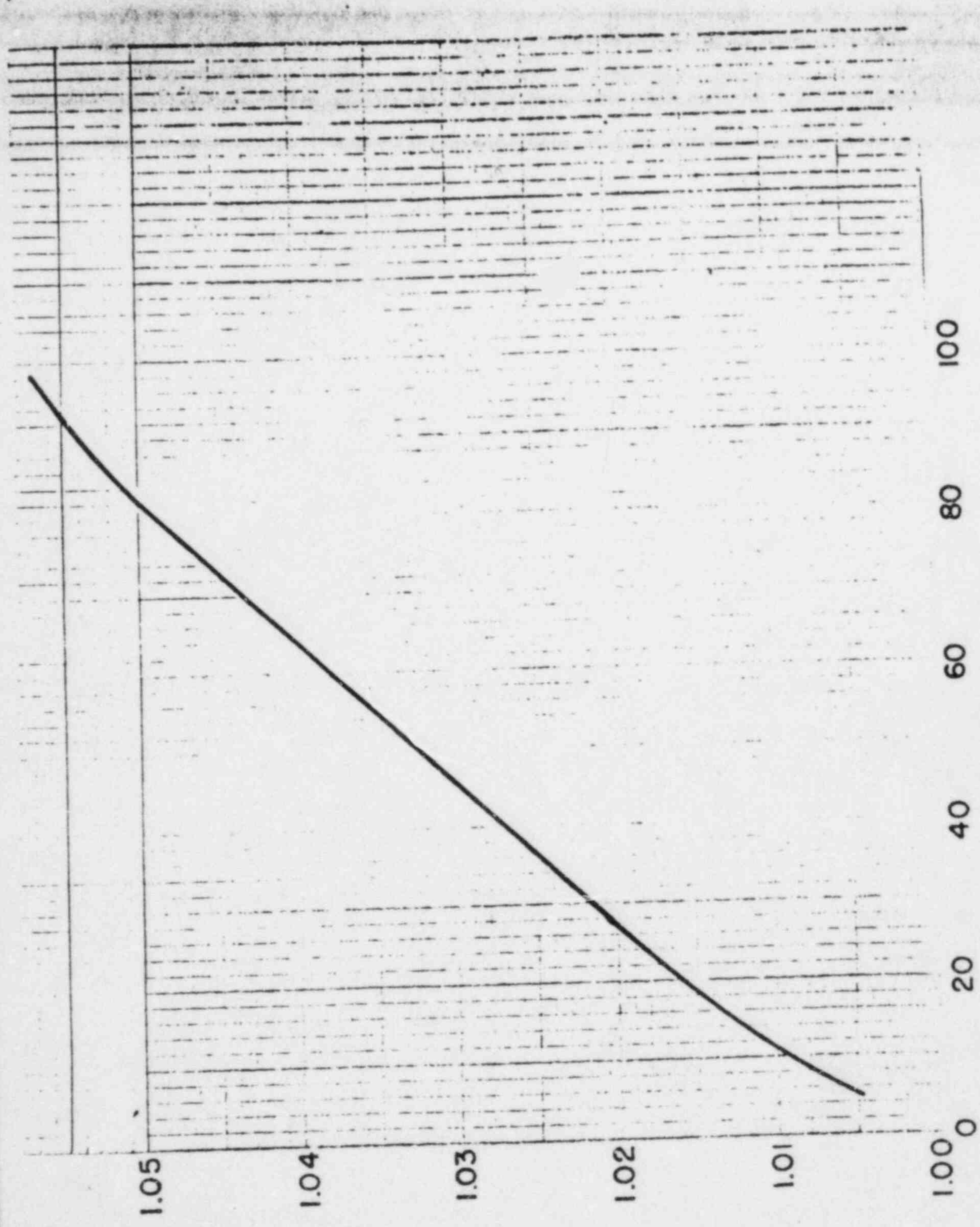
FORT CALHOUN TECHNICAL SPECIFICATIONS	LIMITING CONDITION FOR OPERATION FOR EXCORE MONITORING OF LINEAR HEAT RATES UP TO A MAXIMUM OF 15.22 KW/FT	FIGURE 2-6

Amendment No. 8, 20, 32, 43, 47



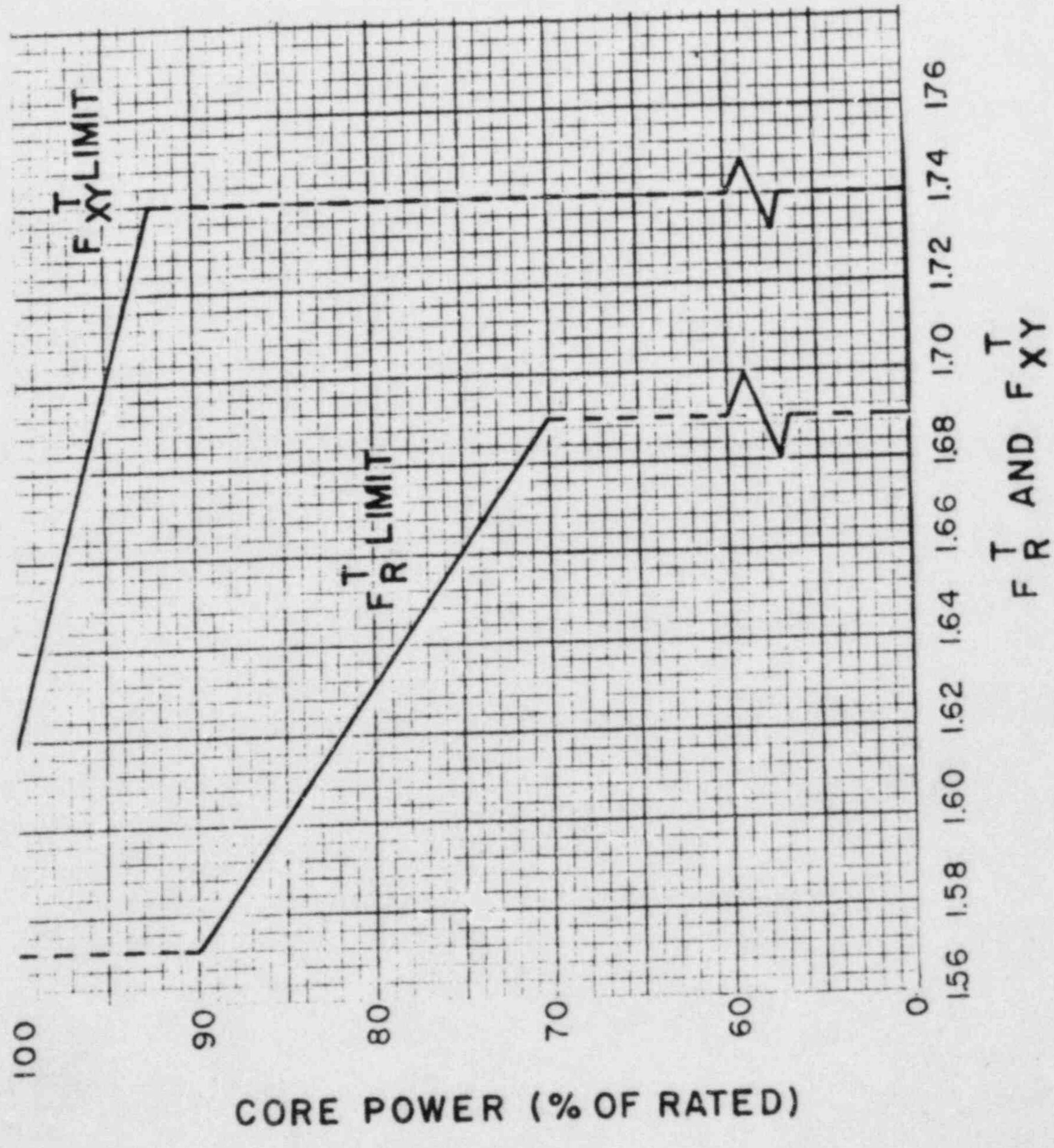
FORT CALHOUN TECHNICAL SPECIFICATIONS	LIMITING CONDITION FOR OPERATION FOR DNB MONITORING	FIGURE 2-7

Amendment No. 8, 20, 32, 47 4



FORT CALHOUN TECHNICAL SPECIFICATIONS	FLUX PEAKING AUGMENTATION FACTORS	FIGURE 2-8

Amendment No. 17



FORT CALHOUN TECHNICAL SPECIFICATIONS	F_R^T, F_{XY}^T, AND CORE POWER LIMITATIONS	FIGURE 2-9

Amendment No. ~~8~~, ~~19~~, ~~22~~, ~~41~~, 47

2.0 LIMITING CONDITIONS FOR OPERATION
2.10 Reactor Core (Continued)
2.10.3 In-Core Instrumentation

(a) An operable incore detector string shall consist of three operable rhodium detectors.

(b) A quadrant symmetric incore detector string location shall consist of a location with a symmetric counterpart in any other quadrant.

(c) The initial measurement of the linear heat rate, F_R^T, F_{xy}^T and azimuthal power tilt shall consist of the first full core power distribution calculation based on incore detector signals made at a power level greater than 40 percent of rated power following each fuel loading.

If an initial measurement of the linear heat rate, F_R^T, F_{xy}^T and azimuthal power tilt cannot be made with an operable incore detector system as defined above, reactor power shall be restricted to less than 75 percent of the peak allowable heat rate.

Basis

The in-core instrument system is used to monitor core performance and to insure operation within the limits used as initial conditions for the safety analysis in three ways:

(1) to verify that the radial peaking factors (F_{xy}^T and F_R^T) are less than the limits specified in Specifications 2.10.4(2) and 2.10.4(3),

(2) to actuate alarms set on each individual instrument to insure operation within specified kw/ft limits of Figure 2-5, and

(3) to determine the axial shape index for periodic verification of the calibration of the ex-core detector system.

The specification requires a minimum number of detectors and proper distribution to perform these functions. In-core rhodium detectors in conjunction with analytical computer codes calculate power distributions from which F_{xy} and F_R are determined. Alarm limits are set on each in-core instrument to insure operation within specified kw/ft limits.

Calibration of the ex-core detector input to the APD calculator is required to eliminate ASI uncertainties due to instrument drift and axially nonuniform detector exposure. If the recalibration is not performed in the period specified, the prescribed steps will assure safe operation of the reactor.

References

(1) Evaluation of Uncertainty in the Nuclear Form Factor Measured by Self-Powered Fixed In-Core Detector Systems - CENPD-153, August, 1974.

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2.10.4 Power Distribution Limits

Applicability

Applies to power operation conditions.

Objective

To ensure that peak linear heat rates, DNB margins, and radial peaking factors are maintained within acceptable limits during power operation.

Specification

(1) Linear Heat Rate

The linear heat rate shall not exceed the limits shown on Figure 2-5 when the following factors are appropriately included:

1. Flux peaking augmentation factors are shown in Figure 2-8,
2. A measurement-calculational uncertainty factor of 1.07,
3. An engineering uncertainty factor of 1.03,
4. A linear heat rate uncertainty factor of 1.002 due to axial fuel densification and thermal expansion, and
5. A power measurement uncertainty factor of 1.02.

(a) When the linear heat rate is continuously monitored by the incore detectors, and the linear heat rate is exceeding its limits as indicated by four or more valid coincident incore detector alarms, either:

(i) Restore the linear heat rate to within its limits within one hour, or

(ii) Be in at least hot standby within the next 6 hours.

(b) If while operating under the provisions of part (a) the plant computer incore detector alarms become inoperable, operation may be continued without reducing power provided each of the following conditions is satisfied:

(i) A core power distribution was obtained utilizing incore detectors within 7 days prior to the incore detector alarm outage and the measured peak linear heat rate was no greater than 90% of the value allowed by (1) above.

(ii) The Axial Shape Index as measured by excore detectors remains within $\pm .05$ of the value obtained at the time of the last measured incore power distribution.

(iii) Power is not increased nor has it been increased since the time of the last incore distribution.

(c) When the linear heat rate is continuously monitored by the excore detectors, withdraw the full length CEA's beyond the long term insertion limits of Specification 2.10.2.7. If the linear heat rate is exceeding its limits as determined by the Axial Shape Index, Y_I, being outside the limits of Figure 2-6, where 100 percent of the allowable power represents the maximum power allowed by the following expression:

$$\frac{L}{15.22} \times M$$

where

1. L is the maximum allowable linear heat rate as determined from Figure 2-5 and is based on the core average burnup at the time of the latest incore power map.

2. M is the maximum allowable fraction of rated thermal power as determined by the F_{xy}^T limit curve of Figure 2-9 when monitoring by excore detectors. M = 1 when monitoring kw/ft using incore detectors.

(i) Restore the reactor power and Axial Shape Index, Y_I, to within the limits of Figure 2-6 within 2 hours, or

(ii) Be in at least hot standby within the next 6 hours.

(2) Total Integrated Radial Peaking Factor

The calculated value of $F_R{}^T$ defined by $F_R{}^T = F_R\ (1+T_q)$ shall be limited to ≤ 1.57. F_R is determined from a power distribution map with no part length CEA's inserted and with all full length CEA's at or above the Long Term Steady State Insertion Limit for the existing Reactor Coolant Pump combination. The azimuthal tilt, T_q, is the measured value of T_q at the time F_R is determined.

With $F_R{}^T > 1.57$ within 6 hours:

(a) Reduce power to bring power and $F_R{}^T$ within the limits of Figure 2-9, withdraw the full length CEA's to or beyond the Long Term Steady State Insertion Limits of Specification 2.10.2(7), and fully withdraw the PLCEA's, or

(b) Be in at least hot standby.

(3) Total Planar Radial Peaking Factor

The calculated value of $F_{xy}{}^T$ defined as $F_{xy}{}^T = F_{xy}\ (1+T_q)$ shall be limited to ≤ 1.62. F_{xy} shall be determined from a power distribution map with no part length CEA inserted and with all full length CEA's at or above the Long Term Steady State Insertion Limit for the existing Reactor Coolant Pump combination. This determination shall be limited to core planes between 15% and 85% of full core height inclusive and shall exclude regions influenced by grid effects. The azimuthal tilt, T_q, is the measured value of T_q at the time F_{xy} is determined.

With $F_{xy}{}^T > 1.62$ within 6 hours:

(a) Reduce power to bring power and $F_{xy}{}^T$ to within the limits of Figure 2-9, and withdraw the full length CEA's to or beyond the Long Term Steady State Insertion Limits of Specification 2.10.2(7) and fully withdraw the PLCEA's, or

(b) Be in at least hot standby.

2.0 LIMITING CONDITIONS FOR OPERATION
2.10 Reactor Core (Continued)
2.10.4 Power Distribution Limits (Continued)

(4) Azimuthal Power Tilt (T_q)

When operating above 70% of rated power, the azimuthal power tilt (T_q) shall not exceed 0.03.

(a) With the indicated azimuthal power tilt determined to be >0.03 but <0.10, correct the power tilt within two hours or determine within the next 6 hours and at least once per subsequent 8 hours, that the total integrated radial peaking factor, F_R^T, is within the limit of Specification 2.10.4(2) and that the total planar radial peaking factor, F_{xy}^T, is within the limit of 2.10.4(3), or reduce power to less than 70% of rated power within 8 hours of confirming T_q >0.03.

(b) With the indicated power tilt determined to be $\geq$.10, power operation may proceed up to 2 hours provided F_R^T and F_{xy}^T do not exceed the power limits of Figure 2-9, or be in at least hot standby within 6 hours. Subsequent operation for the purpose of measurement to identify the cause of the tilt is allowable provided the power level is restricted to 20% of the maximum allowable thermal power level for the existing reactor coolant pump combination.

TABLE 2.6

DNB MARGIN

SPECIFIED OPERATING LIMITS

Monitored Parameter	OPERATING LIMIT 4 Pump
Cold Leg Temperature	$\leq$(545°F)*
Pressurizer Pressure	$\geq$(2075 psia)*
Reactor Coolant Flow	$\geq$(195,700 gpm)**
Axial Shape Index	$\leq$(Figure 2-7)

*Limit not applicable during either a thermal power ramp increase in excess of 5% of rated thermal power per minute or a thermal power step increase of greater than 10% of rated thermal power.

**This number is an actual limit (not including uncertainties). All other values in this table are indicated values and include an allowance for measurement uncertainty (e.g., 545°F, indicated, allows for an actual T_c of 547°F).