



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

PORTLAND GENERAL ELECTRIC COMPANY

THE CITY OF EUGENE, OREGON

PACIFIC POWER AND LIGHT COMPANY

DOCKET NO. 50-344

TROJAN NUCLEAR PLANT

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No.46
License No. NPF-1

1. The Nuclear Regulatory Commission (the Commission) has found that:
 - A. The application for amendment by Portland General Electric Company, the City of Eugene, Oregon, and Pacific Power and Light Company (the licensees) dated January 5, 1979, as supplemented February 22 and November 5, 1979, 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 2.C.(2) of Facility Operating License No. NPF-1 is hereby amended to read as follows:

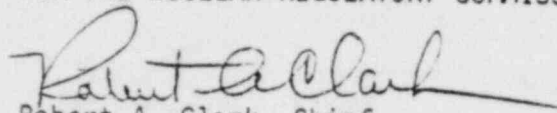
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(2) Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 46, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications, except as noted in paragraphs 2.C.(10) and 2.c.(11) below.

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

FOR THE NUCLEAR REGULATORY COMMISSION



Robert A. Clark, Chief
Operating Reactors Branch #3
Division of Licensing

Attachment:
Changes to the Technical
Specifications

Date of Issuance: July 25, 1980

ATTACHMENT TO LICENSE AMENDMENT NO. 46

FACILITY OPERATING LICENSE NO. NPF-1

DOCKET NO. 50-344

Replace the following pages of the Appendix "A" Technical Specifications with the enclosed pages. The revised pages are identified by Amendment number and contain vertical lines indicating the areas of change. The corresponding overleaf pages are also provided to maintain document completeness.

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2.0 SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

2.1 SAFETY LIMITS

REACTOR CORE

2.1.1 The combination of THERMAL POWER, pressurizer pressure, and the highest operating loop coolant temperature (T_{avg}) shall not exceed the limits shown in Figures 2.1-1 and 2.1-2 for 4 and 3rd loop operation, respectively.

APPLICABILITY: MODES 1 and 2.

ACTION:

Whenever the point defined by the combination of the highest operating loop average temperature and THERMAL POWER has exceeded the appropriate pressurizer pressure line, be in HOT STANDBY within 1 hour.

REACTOR COOLANT SYSTEM PRESSURE

2.1.2 The Reactor Coolant System pressure shall not exceed 2735 psig.

APPLICABILITY: MODES 1, 2, 3, 4 and 5.

ACTION:

MODES 1 and 2

Whenever the Reactor Coolant System pressure has exceeded 2735 psig, be in HOT STANDBY with the Reactor Coolant System pressure within its limit within 1 hour.

MODES 3, 4 and 5

Whenever the Reactor Coolant System pressure has exceeded 2735 psig, reduce the Reactor Coolant System pressure to within its limit within 5 minutes.

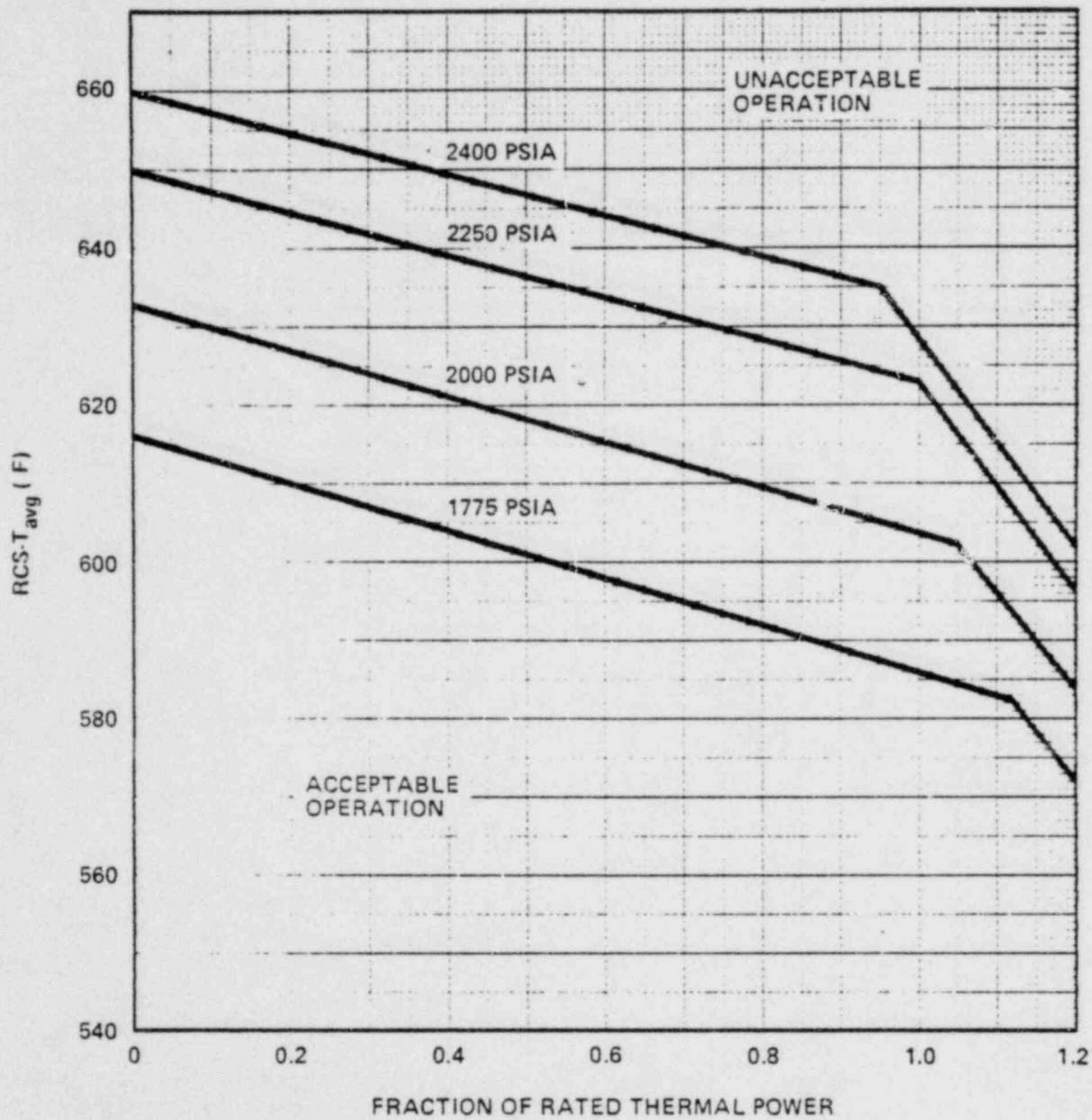


Figure 2.1-1 Reactor Core Safety Limit—Four Loops in Operation.

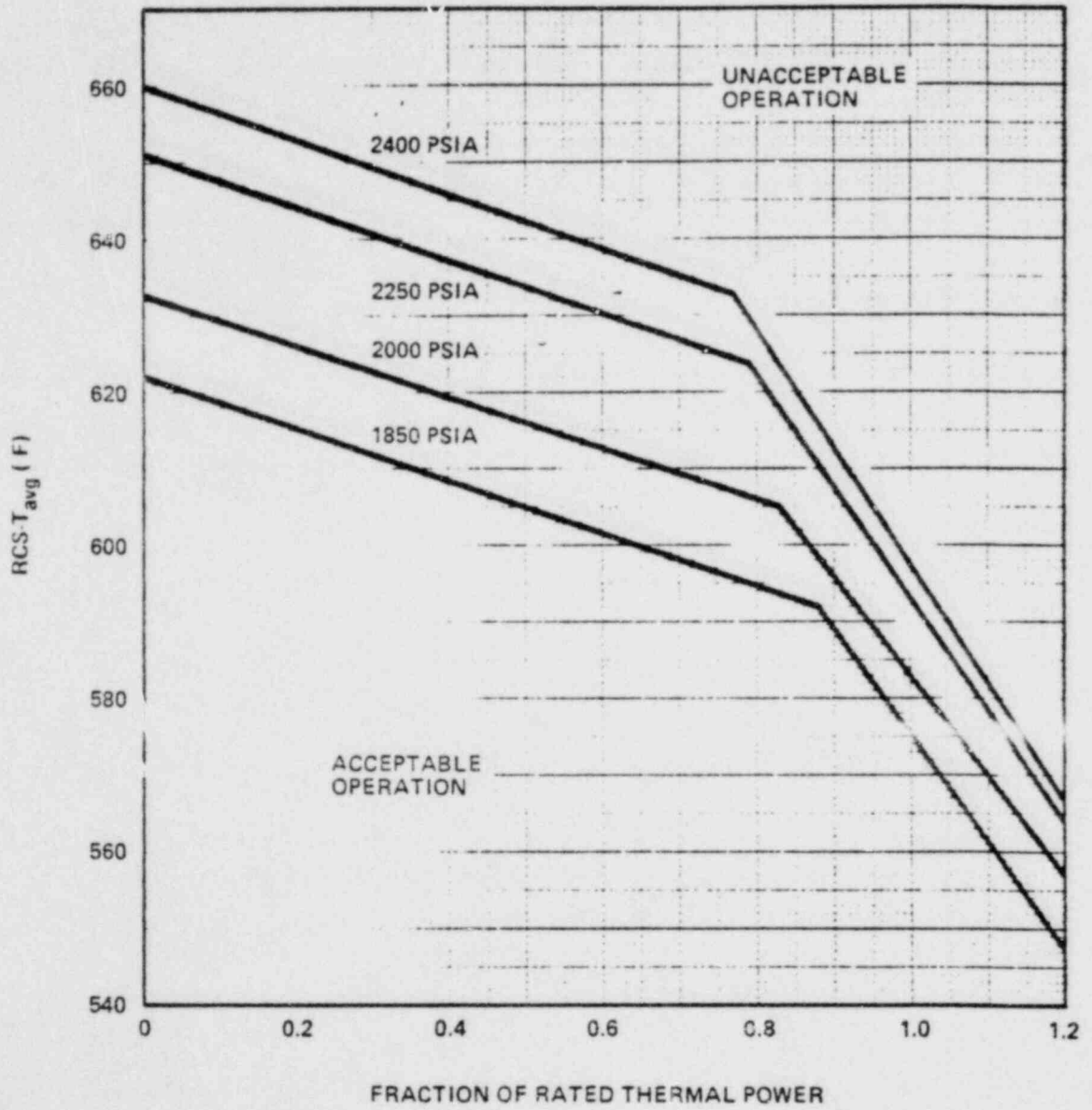


Figure 2.1-2 Reactor Core Safety Limit—Three Loops in Operation.

SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

2.2 LIMITING SAFETY SYSTEM SETTINGS

REACTOR TRIP SYSTEM INSTRUMENTATION SETPOINTS

2.2.1 The reactor trip system instrumentation setpoints shall be set consistent with the Trip Setpoint values shown in Table 2.2-1.

APPLICABILITY: As shown for each channel in Table 3.3-1.

ACTION:

With a reactor trip system instrumentation setpoint less conservative than the value shown in the Allowable Values column of Table 2.2-1, declare the channel inoperable and apply the applicable ACTION statement requirement of Specification 3.3.1.1 until the channel is restored to OPERABLE status with its trip setpoint adjusted consistent with the Trip Setpoint value.

TABLE 2.2-1

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

<u>FUNCTIONAL UNIT</u>	<u>TRIP SETPOINT</u>	<u>ALLOWABLE VALUES</u>
1. Manual Reactor Trip	Not Applicable	Not Applicable
2. Power Range, Neutron Flux	Low Setpoint - $\leq 25\%$ of RATED THERMAL POWER High Setpoint - $\leq 109\%$ of RATED THERMAL POWER	Low Setpoint - $\leq 26\%$ of RATED THERMAL POWER High Setpoint - $\leq 110\%$ of RATED THERMAL POWER
3. Power Range, Neutron Flux, High Positive Rate	$\leq 5\%$ of RATED THERMAL POWER with a time constant ≥ 2 seconds	$\leq 5.5\%$ of RATED THERMAL POWER with a time constant ≥ 2 seconds
4. Power Range, Neutron Flux, High Negative Rate	$\leq 5\%$ of RATED THERMAL POWER with a time constant ≥ 2 seconds	$\leq 5.5\%$ of RATED THERMAL POWER with a time constant ≥ 2 seconds
5. Intermediate Range, Neutron Flux	$\leq 25\%$ of RATED THERMAL POWER	$\leq 30\%$ of RATED THERMAL POWER
6. Source Range, Neutron Flux	$\leq 10^5$ counts per second	$\leq 1.3 \times 10^5$ counts per second
7. Overtemperature ΔT	See Note 1	See Note 3
8. Overpower ΔT	See Note 2	See Note 3
9. Pressurizer Pressure--Low	≥ 1865 psig	≥ 1855 psig
10. Pressurizer Pressure--High	≤ 2385 psig	≤ 2395 psig
11. Pressurizer Water Level--High	$\leq 92\%$ of instrument span	$\leq 93\%$ of instrument span
12. Loss of Flow	$\geq 90\%$ of design flow per loop*	$\geq 89\%$ of design flow per loop*

*Design Flow is 92,925 gpm per loop.

TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

<u>FUNCTIONAL UNIT</u>	<u>TRIP SETPOINT</u>	<u>ALLOWABLE VALUES</u>
13. Steam Generator Water Level--Low-Low	\geq 5% of narrow range instrument span--each steam generator	\geq 4% of narrow range instrument span--each steam generator
14. Steam/Feedwater Flow Mismatch and Low Steam Generator Water Level	$<$ 40% of full steam flow at RATED THERMAL POWER coincident with steam generator water level \geq 25% of narrow range instrument span--each steam generator	$<$ 42.5% of full steam flow at RATED THERMAL POWER coincident with steam generator water level \geq 24% of narrow range instrument span--each steam generator
15. Undervoltage-Reactor Coolant Pumps	\geq 68% each bus	\geq 67% each bus
16. Underfrequency-Reactor Coolant Pumps	\geq 57.5 Hz - each bus	\geq 57.4 Hz - each bus
17. Turbine Trip		
A. Low Trip System Pressure	\geq 800 psig	\geq 700 psig
B. Turbine Stop Valve Closure	\geq 1% open	\geq 1% open
18. Safety Injection Input from ESF	Not Applicable	Not Applicable
19. Reactor Coolant Pump Breaker Position Trip	Not Applicable	Not Applicable

TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTSNOTATION

NOTE 1: Overtemperature $\Delta T \leq \Delta T_0 \left[K_1 - K_2 \left(\frac{1 + \tau_1 S}{1 + \tau_2 S} \right) (T - T') + K_3 (P - P') - f_1 (\Delta I) \right]$

where: ΔT_0 = Indicated ΔT at RATED THERMAL POWER (set to 100% of RATED THERMAL POWER when $\Delta T = 61.45^\circ\text{F}$)

T = Average temperature, $^\circ\text{F}$

$T' \leq 584.8^\circ\text{F}$ (indicated T_{avg} at RATED THERMAL POWER)

P = Pressurizer pressure, psig

$P' = 2235$ psig (indicated RCS nominal operating pressure)

$\frac{1 + \tau_1 S}{1 + \tau_2 S}$ = The function generated by the lead-lag controller for T_{avg} dynamic compensation

τ_1 & τ_2 = Time constants utilized in the lead-lag controller for T_{avg} $\tau_1 = 30$ secs,
 $\tau_2 = 4$ secs.

S = Laplace transform operator

Operation with 4 Loops

$$K_1 = 1.32$$

$$K_2 = 0.02109 \text{ per } ^\circ\text{F}$$

$$K_3 = 0.00100 \text{ per psi}$$

Operation with 3 Loops

$$K_1 = 1.17$$

$$K_2 = 0.02109 \text{ per } ^\circ\text{F}$$

$$K_3 = 0.00100 \text{ per psi}$$

TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTSNOTATION (Continued)

and $f_1(\Delta I)$ is a function of the indicated difference between top and bottom detectors of the power-range nuclear ion chambers; with gains to be selected based on measured instrument response during plant startup tests such that:

- (i) for $q_t - q_b$ between - 45 percent and + 3 percent, $f_1(\Delta I) = 0$ (where q_t and q_b are percent RATED THERMAL POWER in the top and bottom halves of the core respectively, and $q_t + q_b$ is total THERMAL POWER in percent of RATED THERMAL POWER).
- (ii) for each percent that the magnitude of $(q_t - q_b)$ exceeds - 45 percent, the ΔT trip setpoint shall be automatically reduced by 2.20 percent of its value at RATED THERMAL POWER.
- (iii) for each percent that the magnitude of $(q_t - q_b)$ exceeds + 3 percent, the ΔT trip setpoint shall be automatically reduced by 2.66 percent of its value at RATED THERMAL POWER.

Note 2: Overpower $\Delta T \leq \Delta T_0 \left[K_4 - K_5 \left(\frac{\tau_3 S}{+\tau_3 S} \right) T - K_6 (T - T'') - f_2(\Delta I) \right]$

Where: ΔT_0 = Indicated ΔT at rated power

T = Average temperature, °F

T'' = Indicated T_{avg} at RATED THERMAL POWER ≤ 584.8 °F

K_4 = 1.08

K_5 = 0.02/°F for increasing average temperature

K_6 = 0.00137/°F for $T > T''$; $K_6 = 0$ for $T \leq T''$

TABLE 2.2-1 (Continued)REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTSNOTATION (Continued)

$\frac{\tau_3 S}{1 + \tau_3 S}$ = The function generated by the rate lag controller for T_{avg}
dynamic compensation

τ_3 = Time constant utilized in the rate lag controller for T_{avg}
 $\tau_3 = 10$ secs.

S = Laplace transform operator

$f_2(\Delta I) = 0$ for all ΔI

NOTE 3: The channel's maximum trip point shall not exceed its computed trip point by more than 2 percent.

2.1 SAFETY LIMITS

BASES

2.1.1 REACTOR CORE

The restrictions of this safety limit prevent overheating of the fuel and possible cladding perforation which would result in the release of fission products to the reactor coolant. Overheating of the fuel cladding is prevented by restricting fuel operation to within the nucleate boiling regime where the heat transfer coefficient is large and the cladding surface temperature is slightly above the coolant saturation temperature.

Operation above the upper boundary of the nucleate boiling regime could result in excessive cladding temperatures because of the onset of departure from nucleate boiling (DNB) and the resultant sharp reduction in heat transfer coefficient. DNB is not a directly measurable parameter during operation and therefore THERMAL POWER and Reactor Coolant Temperature and Pressure have been related to DNB. This relation has been developed to predict the DNB flux and the location of DNB for axially uniform and non-uniform heat flux distributions. The local DNB heat flux ratio, DNBR, defined as the ratio of the heat flux that would cause DNB at a particular core location to the local heat flux, is indicative of the margin to DNB.

The DNB design basis is that there must be at least a 95 percent probability with 95 percent confidence that DNB will not occur when the minimum DNBR is at the DNBR limit.

In meeting this design basis, uncertainties in plant operating parameters, nuclear and thermal parameters, and fuel fabrication parameters are considered statistically such that there is at least a 95 percent confidence that the minimum DNBR for the limiting rod is greater than or equal to the DNBR limit. The uncertainties in the above plant parameters are used to determine the plant DNBR uncertainty. This DNBR uncertainty, combined with the correlation DNBR limit, establishes a design DNBR value which must be met in plant safety analyses using values of input parameters without uncertainties.

The curves of Figures 2.1-1 and 2.1-2 show the loci of points of THERMAL POWER, Reactor Coolant System pressure, and average temperature for which the calculated DNBR is no less than the design DNBR value or the average enthalpy at the vessel exit is less than the enthalpy of saturated liquid.

SAFETY LIMITS

BASES

The curves are based on an enthalpy hot channel factor, $F_{\Delta H}^N$, of 1.49 and a reference cosine with a peak of 1.55 $F_{\Delta H}^N$ for axial power shape. An allowance is included for an increase in $F_{\Delta H}^N$ at reduced power based on the expression:

$$F_{\Delta H}^N = 1.49 [1 + 0.2 (1-P)]$$

where P is the fraction of RATED THERMAL POWER

These limiting heat flux conditions are higher than those calculated for the range of all control rods fully withdrawn to the maximum allowable control rod insertion assuming the axial power imbalance is within the limits of the $f(\Delta I)$ function of the Overtemperature ΔT trip. When the axial power imbalance is not within the tolerance, the axial power imbalance effect on the Overtemperature ΔT trips will reduce the set-points to provide protection consistent with core safety limits.

2.1.2 REACTOR COOLANT SYSTEM PRESSURE

The restriction of this Safety Limit protects the integrity of the Reactor Coolant System from overpressurization and thereby prevents the release of radionuclides contained in the reactor coolant from reaching the containment atmosphere.

The reactor pressure vessel and pressurizer are designed to Section III of the ASME Code for Nuclear Power Plant which permits a maximum transient pressure of 110% (2735 psig) of design pressure. The Reactor Coolant System piping, valves and fittings, are designed to ANSI B 31.7-1969, which permits a maximum transient pressure of 120% (2985 psig) of component design pressure. The Safety Limit of 2735 psig is therefore consistent with the design criteria and associated code requirements.

The entire Reactor Coolant System is hydrotested at 3107 psig to demonstrate integrity prior to initial operation.

2.2 LIMITING SAFETY SYSTEM SETTINGS

BASES

2.2.1 REACTOR TRIP SYSTEM INSTRUMENTATION SETPOINTS

The Reactor Trip Setpoint Limits specified in Table 2.2-1 are the values at which the Reactor Trips are set for each parameter. The Trip Setpoints have been selected to ensure that the reactor core and reactor coolant system are prevented from exceeding their safety limits. Operation with a trip set less conservative than its Trip Setpoint but within its specified Allowable Value is acceptable on the basis that each Allowable Value is equal to or less than the drift allowance assumed for each trip in the safety analyses.

Manual Reactor Trip

The Manual Reactor Trip is a redundant channel to the automatic protective instrumentation channels and provides manual reactor trip capability.

Power Range, Neutron Flux

The Power Range, Neutron Flux channel high setpoint provides reactor core protection against reactivity excursions which are too rapid to be protected by temperature and pressure protective circuitry. The low setpoint provides redundant protection in the power range for a power excursion beginning from low power. The trip associated with the low setpoint may be manually bypassed when P-10 is active (two of the four power range channels indicate a power level of above approximately 10 percent of RATED THERMAL POWER) and is automatically reinstated when P-10 becomes inactive (three of the four channels indicate a power level below approximately 10 percent of RATED THERMAL POWER).

Power Range, Neutron Flux, High Rates

The Power Range Positive Rate trip provides protection against rapid flux increases which are characteristic of rod ejection events from any power level. Specifically, this trip complements the Power Range Neutron Flux High and Low trips to ensure that the criteria are met for rod ejection from partial power.

LIMITING SAFETY SYSTEM SETTINGS

BASES

The Power Range Negative Rate trip provides protection to ensure that the minimum DNBR is maintained above the design DNBR value for multiple control rod drop accidents. The analysis of a single control rod drop accident indicates a return to full power may be initiated by the automatic control system in response to a continued full power turbine load demand or by the negative moderator temperature feedback. This transient will not result in a calculated DNBR of less than the design DNBR value, therefore single rod drop protection is not required.

Intermediate and Source Range, Nuclear Flux

The Intermediate and Source Range, Nuclear Flux trips provide reactor core protection during reactor startup. These trips provide redundant protection to the low setpoint trip of the Power Range, Neutron Flux channels. The Source Range Channels will initiate a reactor trip at about 10^{+5} counts per second unless manually blocked when P-6 becomes active. The Intermediate Range Channels will initiate a reactor trip at a current level proportional to approximately 25 percent of RATED THERMAL POWER unless manually blocked when P-10 becomes active. No credit was taken for operation of the trips associated with either the Intermediate or Source Range Channels in the accident analyses; however, their functional capability at the specified trip settings is required by this specification to enhance the overall reliability of the Reactor Protection System.

Overtemperature ΔT

The Overtemperature ΔT trip provides core protection to prevent DNB for all combinations of pressure, power, coolant temperature, and axial power distribution, provided that the transient is slow with respect to piping transit delays from the core to the temperature detectors (about 4 seconds), and pressure is within the range between the High and Low Pressure reactor trips. This setpoint includes corrections for changes in density and heat capacity of water with temperature and dynamic compensation for piping delays from the core to the loop temperature detectors. With normal axial power distribution, this reactor trip limit is always below the core safety limit as shown in Figure 2.1-1. If axial peaks are greater than design, as indicated by the difference between top and bottom power range nuclear detectors, the reactor trip is automatically reduced according to the notations in Table 2.2-1.

LIMITING SAFETY SYSTEM SETTINGS

BASES

Operation with a reactor coolant loop out of service below the 4 loop P-8 set point does not require reactor protection system set point modification because the P-8 set point and associated trip will prevent DNB during 3 loop operation exclusive of the Overtemperature ΔT set point. Three loop operation above the 4 loop P-8 set point is permissible after resetting the K1, K2 and K3 inputs to the Overtemperature ΔT channels and raising the P-8 set point to its 3 loop value. In this mode of operation, the P-8 interlock and trip functions as a High Neutron Flux trip at the reduced power level.

Overpower ΔT

The Overpower ΔT reactor trip provides assurance of fuel integrity, e.g., no melting, under all possible overpower conditions, limits the required range for Overtemperature ΔT protection, and provides a backup to the High Neutron Flux trip. The setpoint includes corrections for changes in density and heat capacity of water with temperature, and dynamic compensation for piping delays from the core to the loop temperature detectors. No credit was taken for operation of this trip in the accident analyses; however, its functional capability at the specified trip setting is required by this specification to enhance the overall reliability of the Reactor Protection System.

Pressurizer Pressure

The Pressurizer High and Low Pressure trips are provided to limit the pressure range in which reactor operation is permitted. The High Pressure trip is backed up by the pressurizer code safety valves for RCS overpressure protection, and is therefore set lower than the set pressure for these valves (2485 psig). The Low Pressure trip provides protection by tripping the reactor in the event of a loss of reactor coolant pressure.

Pressurizer Water Level

The Pressurizer High Water Level trip ensures protection against Reactor Coolant System overpressurization by limiting the water level to a volume sufficient to retain a steam bubble and prevent water relief

LIMITING SAFETY SYSTEM SETTINGS

BASES

through the pressurizer safety valves. No credit was taken for operation of this trip in the accident analyses; however, its functional capability at the specified trip setting is required by this specification to enhance the overall reliability of the Reactor Protection System.

Loss of Flow

The Loss of Flow trips provide core protection to prevent DNB in the event of a loss of one or more reactor coolant pumps.

Above 10 percent of RATED THERMAL POWER, an automatic reactor trip will occur if the flow in any two loops drops below 90% of nominal full loop flow. Above 38% (P-8) of RATED THERMAL POWER, an automatic reactor trip will occur if the flow in any single loop drops below 90% of nominal full loop flow. This latter trip will prevent the minimum value of the DNBR from going below 1.73 during normal operational transients and anticipated transients when 3 loops are in operation and the Over-temperature ΔT trip setpoint is adjusted to the value specified for all loops in operation. With the Overtemperature ΔT trip setpoint adjusted to the value specified for 3 loop operation, the P-8 trip at 75% RATED THERMAL POWER will prevent the minimum value of the DNBR from going below 1.73 during normal operational transients and anticipated transients with 3 loops in operation.

Steam Generator Water Level

The Steam Generator Water Level Low-Low trip provides core protection by preventing operation with the steam generator water level below the minimum volume required for adequate heat removal capacity. The specified setpoint provides allowance that there will be sufficient water inventory in the steam generators at the time of trip to allow for starting delays of the auxiliary feedwater system.

Steam/Feedwater Flow Mismatch and Low Steam Generator Water Level

The Steam/Feedwater Flow Mismatch in coincidence with a Steam Generator Low Water Level trip is not used in the transient and accident analyses but is included in Table 2.2-1 to ensure the functional capability of the specified trip settings and thereby enhance the overall

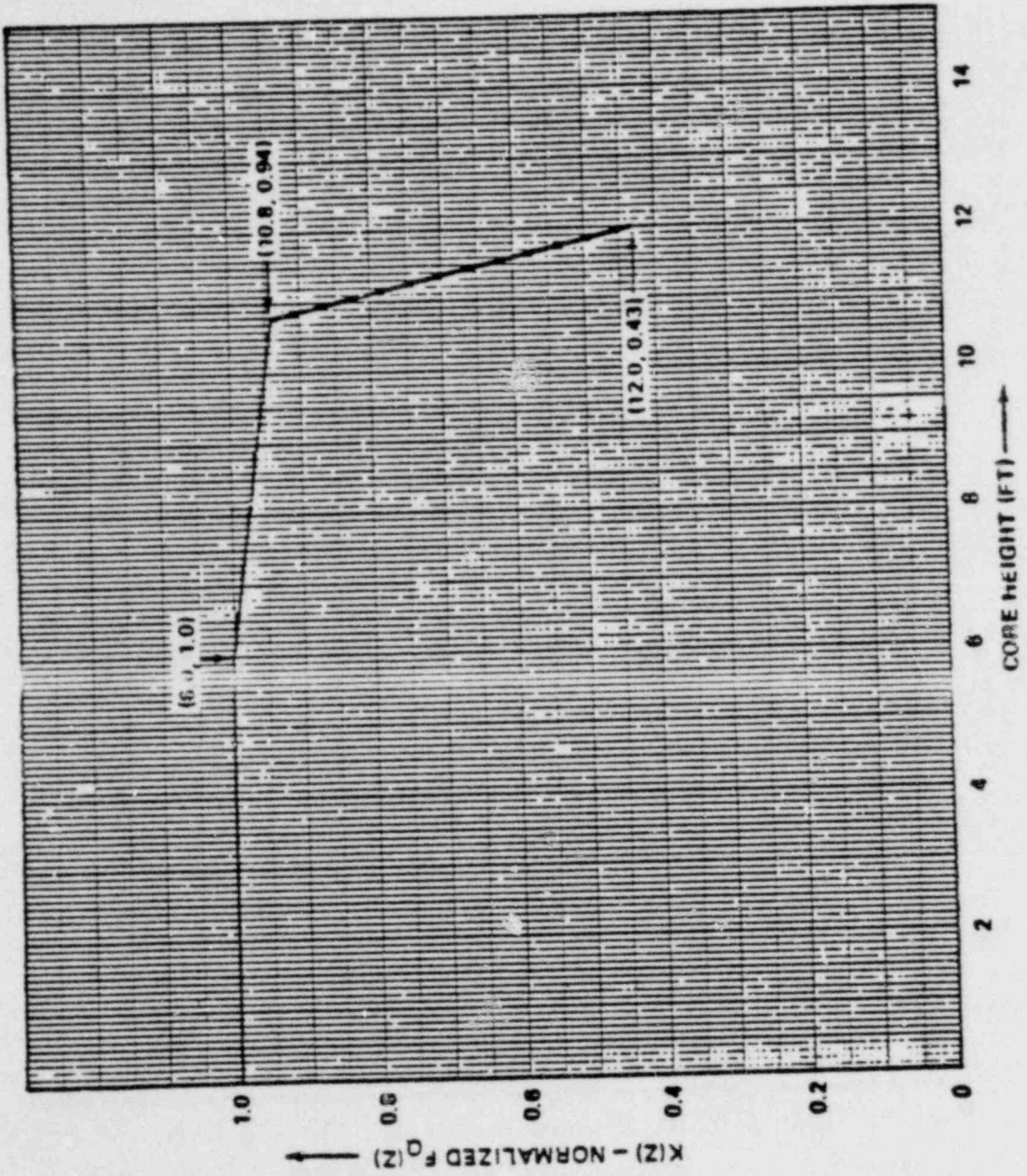


FIGURE 3.2.2
 $K(Z)$ - NORMALIZED $F_0(Z)$ AS A FUNCTION
 OF CORE HEIGHT

POWER DISTRIBUTION LIMITS

RCS FLOWRATE AND F_R

LIMITING CONDITION FOR OPERATION

3.2.3 The combination of indicated Reactor Coolant System (RCS) total flow rate and F_R shall be maintained within the region of allowable operation (above and to the left of the line) shown on Figures 3.2-3 and 3.2-4 for 4- and 3-loop operation, respectively.

Where:

a.
$$F_R = \frac{F_{\Delta H}^N}{1.49 (1.0 + 0.2 (1.0 - P))}$$
, and

b.
$$P = \frac{\text{THERMAL POWER}}{\text{RATED THERMAL POWER}}$$

APPLICABILITY: MODE 1

ACTION:

With the combination of RCS total flow rate and F_R outside the region of acceptable operation shown on Figure 3.2-3 or 3.2-4 (as applicable):

- a. Within 2 hours:
 1. Either restore the combination of RCS flow rate and F_R to within the above limits, or
 2. Reduce THERMAL POWER to less than 50% of RATED THERMAL POWER and reduce the Power Range Neutron Flux - High trip setpoint to \leq 55% of RATED THERMAL POWER within the next 4 hours.
- b. Within 24 hours of initially being outside the above limits, verify through incore flux mapping and RCS total flow rate comparison that the combination of F_R and RCS total flow rate are restored to within the above limits, or reduce THERMAL POWER to less than 5% of RATED THERMAL POWER within the next 2 hours.

POWER DISTRIBUTION LIMITS

ACTION: (Continued)

- c. Identify and correct the cause of the out-of-limit condition prior to increasing THERMAL POWER above the reduced THERMAL POWER limit required by ACTION items a.2 and/or b above; subsequent POWER OPERATION may proceed provided that the combination of R and indicated RCS total flow rate are demonstrated, through incore flux mapping and RCS total flow rate comparison, to be within the region of acceptable operation shown on Figure 3.2-3 or 3.2-4 (as applicable) prior to exceeding the following THERMAL POWER levels:
1. A nominal 50% of RATED THERMAL POWER,
 2. A nominal 75% of RATED THERMAL POWER, and
 3. Within 24 hours of attaining \geq 95% of RATED THERMAL POWER.

SURVEILLANCE REQUIREMENTS

4.2.3.1 The provisions of Specification 4.0.4 are not applicable.

4.2.3.2 The combination of indicated RCS total flow rate and F_D shall be determined to be within the region of acceptable operation of Figure 3.2-3 or 3.2-4 (as applicable):

- a. Prior to operation above 75% of RATED THERMAL POWER after each fuel loading, and
- b. At least once per 3] Effective Full Power Days.

Where:

$$F_R = \frac{F_{\Delta H}^N}{1.49 \{1.0 + 0.2 (1.0 - P)\}}, \text{ and}$$

$F_{\Delta H}^N$ = Measured values of $F_{\Delta H}^N$ obtained by using the movable incore detectors to obtain a power distribution map. The measured values of $F_{\Delta H}^N$ shall be used to calculate F_R since Figures 3.2-3 and 3.2-4 include measurement calculational uncertainties of 3.5% for flow and 4% for incore measurement of $F_{\Delta H}^N$.

4.2.3.3 The RCS total flow rate indicators shall be subjected to a CHANNEL CALIBRATION at least once per 18 months.

4.2.3.4 The RCS total flow rate shall be determined by measurement at least once per 18 months.

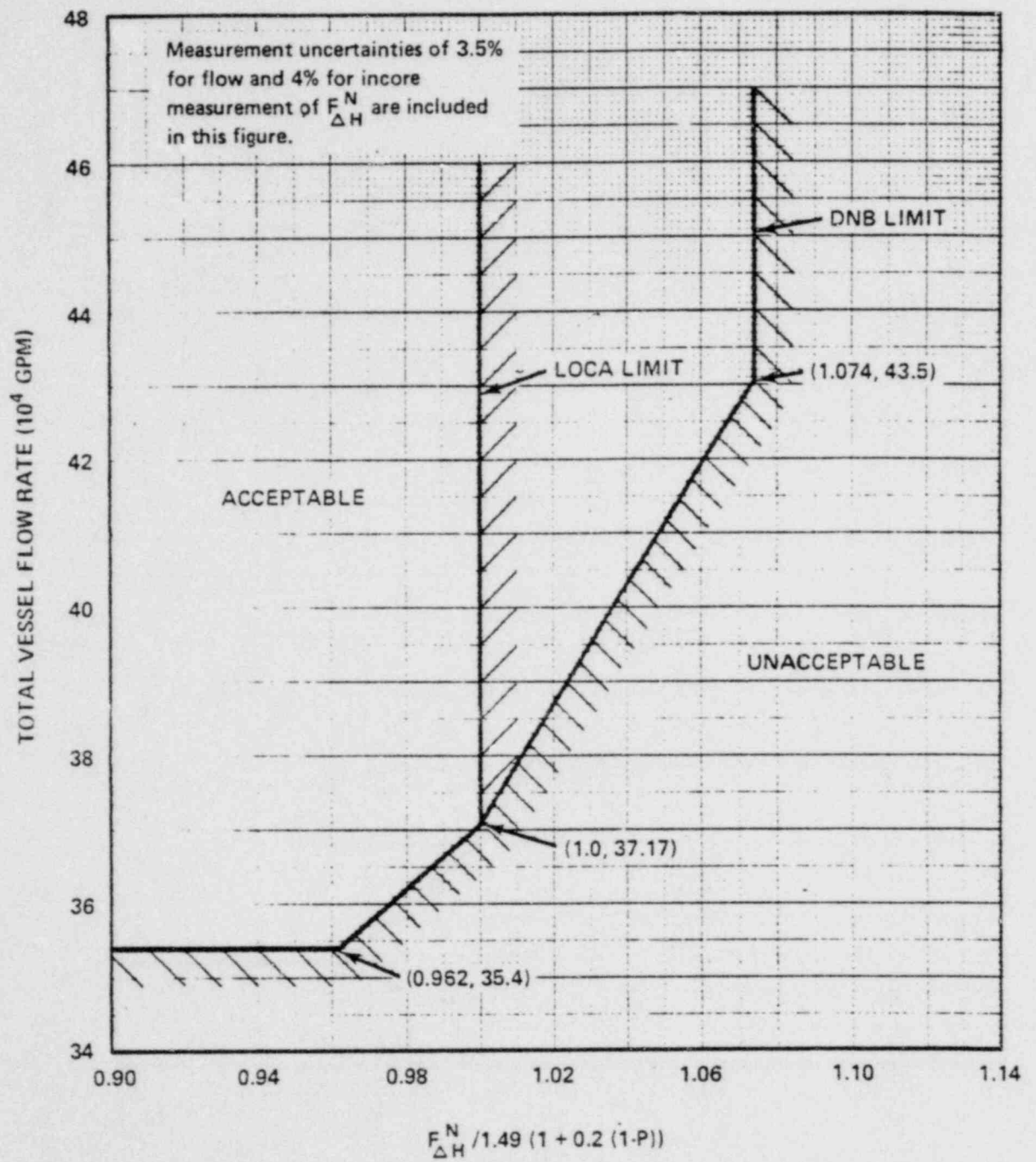


Figure 3.2-3 Flow vs. $F_{\Delta H}^N$ Limit for 4 Loops in Operation.

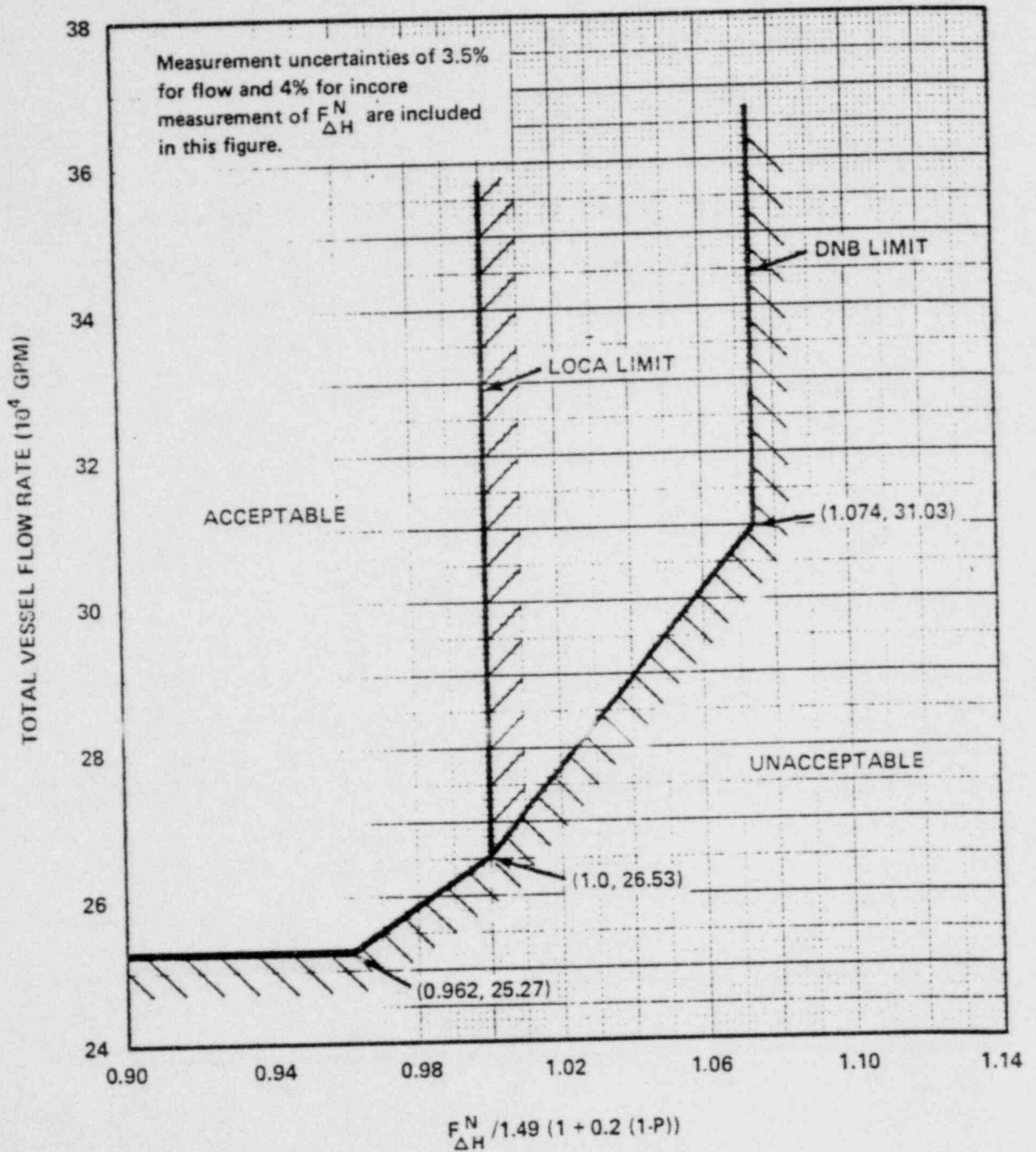


Figure 3.2-4 Flow vs. $F_{\Delta H}^N$ Limit for 3 Loops in Operation.

POWER DISTRIBUTION LIMITS

QUADRANT POWER TILT RATIO

LIMITING CONDITION FOR OPERATION

3.2.4 THE QUADRANT POWER TILT RATIO shall not exceed 1.02.

APPLICABILITY: MODE 1 ABOVE 50% OF RATED THERMAL POWER*

ACTION:

- a. With the QUADRANT POWER TILT RATIO determined to exceed 1.02 but ≤ 1.09 :
 1. Within 2 hours:
 - a) Either reduce the QUADRANT POWER TILT RATIO to within its limit, or
 - b) Reduce THERMAL POWER at least 3% for each 1% of indicated QUADRANT POWER TILT RATIO in excess of 1.0 and similarly reduce the Power Range Neutron Flux-High Trip Setpoints within the next 4 hours.
 2. Verify that the QUADRANT POWER TILT RATIO is within its limit within 24 hours after exceeding the limit or reduce THERMAL POWER to less than 50% of RATED THERMAL POWER within the next 2 hours and reduce the Power Range Neutron Flux-High Trip setpoints to $\leq 55\%$ of RATED THERMAL POWER within the next 4 hours.
 3. Identify and correct the cause of the out of limit condition prior to increasing THERMAL POWER; subsequent POWER OPERATION above 50% of RATED THERMAL power may proceed provided that the QUADRANT POWER TILT RATIO is verified within its limit at least once per hour until verified acceptable at 95% or greater RATED THERMAL POWER.
- b. With the QUADRANT POWER TILT RATIO determined to exceed 1.09 due to misalignment of either a shutdown, control or part length rod:
 1. Reduce THERMAL POWER at least 3% for each 1% of indicated QUADRANT POWER TILT RATIO in excess of 1.0, within 30 minutes.

*See Special Test Exception 3.10.2.

POWER DISTRIBUTION

LIMITING CONDITION FOR OPERATION (Continued)

2. Verify that the QUADRANT POWER TILT RATIO is within its limit within 2 hours after exceeding the limit or reduce THERMAL POWER to less than 50% of RATED THERMAL POWER within the next 2 hours and reduce the Power Range Neutron Flux-High trip Setpoints to $\leq 55\%$ of RATED THERMAL POWER within the next 4 hours.
 3. Identify and correct the cause of the out of limit condition prior to increasing THERMAL POWER; subsequent POWER OPERATION above 50% of RATED THERMAL POWER may proceed provided that the QUADRANT POWER TILT RATIO is verified within its limit at least once per hour until verified acceptable at 95% or greater RATED THERMAL POWER.
- c. With the QUADRANT POWER TILT RATIO determined to exceed 1.09 due to causes other than the misalignment of either a shut-down, control or part length rod:
1. Reduce THERMAL POWER to less than 50% of RATED THERMAL POWER within 2 hours and reduce the Power Range Neutron Flux-High Trip Setpoints to $\leq 55\%$ of RATED THERMAL POWER within the next 4 hours.
 2. Identify and correct the cause of the out of limit condition prior to increasing THERMAL POWER; subsequent POWER OPERATION above 50% of RATED THERMAL POWER may proceed provided that the QUADRANT POWER TILT RATIO is verified within its limit at least once per hour until verified at 95% or greater RATED THERMAL POWER.

SURVEILLANCE REQUIREMENTS

4.2.4 The QUADRANT POWER TILT RATIO shall be determined to be within the limit above 50% of RATED THERMAL POWER by:

- a. Calculating the ratio at least once per 7 days when the alarm is OPERABLE.
- b. Calculating the ratio at least once per 12 hours during steady state operation when the alarm is inoperable.
- c. Using the movable incore detectors to determine the QUADRANT POWER TILT RATIO at least once per 12 hours when one Power Range Channel is inoperable & THERMAL POWER is $> 75\%$ of RATED THERMAL POWER.

POWER DISTRIBUTION LIMITS

DNB PARAMETERS

LIMITING CONDITION FOR OPERATION

3.2.5 The following DNB related parameters shall be maintained within the limits shown on Table 3.2.1:

- a. Reactor Coolant System T_{avg}
- b. Pressurizer Pressure
- c. Reactor Coolant System Total Flow Rate

APPLICABILITY: MODE 1

ACTION:

With any of the above parameters exceeding its limit, restore the parameter to within its limit within 2 hours or reduce THERMAL POWER to less than 5% of RATED THERMAL POWER within the next 4 hours.

SURVEILLANCE REQUIREMENTS

4.2.5 Each of the parameters of Table 3.2-1 shall be verified to be within their limits at least once per 12 hours.

TABLE 3.2-1

DNB PARAMETERSLIMITS

<u>PARAMETER</u>	<u>4 Loops In Operation</u>	<u>3 Loops In Operation</u>
Reactor Coolant System T_{avg}	$\leq 589^{\circ}\text{F}$	$\leq 580.4^{\circ}\text{F}$
Pressurizer Pressure	$\geq 2220 \text{ psia}^*$	$\geq 2220 \text{ psia}^*$
Reactor Coolant System Total Flow Rate	Figure 3.2-3	Figure 3.2-4

*Limit not applicable during either a THERMAL POWER ramp increase in excess of 5% RATED THERMAL POWER per minute or a THERMAL POWER step increase in excess of 10% RATED THERMAL POWER.

TABLE 3.3-1 (Continued)

- a. Less than or equal to 5% of RATED THERMAL POWER, place the inoperable channel in the tripped condition within 1 hour and restore the inoperable channel to operable status within 24 hours after increasing THERMAL POWER above 5% of RATED THERMAL POWER; otherwise, reduce THERMAL POWER to less than 5% of RATED THERMAL POWER within the following 6 hours.
 - b. Above 5% of RATED THERMAL POWER, place the inoperable channel in the tripped condition within 1 hour; operation may continue until performance of the next required CHANNEL FUNCTIONAL TEST.
- ACTION 8 - With the number of OPERABLE channels one less than the Total Numbers of Channels and with the THERMAL POWER level above P-7, place the inoperable channel in the tripped condition within 1 hour; operation may continue until performance of the next required CHANNEL FUNCTIONAL TEST.
- ACTION 9 - With a channel associated with an operating loop inoperable, restore the inoperable channel to OPERABLE status within 2 hours or be in HOT STANDBY within the next 6 hours; however, one channel associated with an operating loop may be bypassed for up to 2 hours for surveillance testing per Specification 4.3.1.1.1.
- ACTION 10 - With one channel inoperable, restore the inoperable channel to OPERABLE status within 2 hours or reduce THERMAL POWER to below P-8 within the next 2 hours. Operation below P-8 may continue pursuant to ACTION 11.
- ACTION 11 - With less than the Minimum Number of Channels OPERABLE, operation may continue provided the inoperable channel is placed in the tripped condition within 1 hour.
- ACTION 12 - With the number of channels OPERABLE one less than required by the Minimum Channels OPERABLE requirement, restore the inoperable channel to OPERABLE status within 48 hours or be in HOT STANDBY within the next 6 hours and/or open the reactor trip breakers.

REACTOR TRIP SYSTEM INTERLOCKS

<u>DESIGNATION</u>	<u>CONDITION AND SETPOINT</u>	<u>FUNCTION</u>
P-6	With 2 of 2 Intermediate Range Neutron Flux Channels $< 6 \times 10^{-11}$ amps.	Prevents or defeats the manual block of source range reactor trip.

TABLE 3.3-1 (Continued)

<u>DESIGNATION</u>	<u>CONDITION AND SETPOINT</u>	<u>FUNCTION</u>
P-7	With 2 of 4 Power Range Neutron Flux Channels \geq 11% of RATED THERMAL POWER or 1 of 2 Turbine impulse chamber pressure channels \geq 66 psia.	Prevents or defeats the automatic block of reactor trip on: Low flow in more than one primary coolant loop, reactor coolant pump under-voltage and under-frequency, turbine trip, pressurizer low pressure, and pressurizer high level.
P-8	With 2 of 4 Power Range Neutron Flux channels \geq 39% of RATED THERMAL POWER.	Prevents or defeats the automatic block of reactor trip on low coolant flow in a single loop.
P-10	With 3 of 4 Power range neutron flux channels $<$ 9% of RATED THERMAL POWER.	Prevents or defeats the manual block of: Power range low setpoint reactor trip, Intermediate range reactor trip, and intermediate range rod stops. Provides input to P-7.

3/4.2 POWER DISTRIBUTION LIMITS

BASES

The specifications of this section provide assurance of fuel integrity during Condition I (Normal Operation) and II (Incidents of Moderate Frequency) events by: (a) maintaining the calculated DNBR in the core at or above design during normal operation and in short term transients, and (b) limiting the fission gas release, fuel pellet temperature & cladding mechanical properties to within assumed design criteria. In addition, limiting the peak linear power density during Condition I events provides assurance that the initial conditions assumed for the LOCA analyses are met and the ECCS acceptance criteria limit of 2200°F is not exceeded.

The definitions of certain hot channel and peaking factors as used in these specifications are as follows:

- $F_Q(Z)$ Heat Flux Hot Channel Factor, is defined as the maximum local heat flux on the surface of a fuel rod at core elevation Z divided by the average fuel rod heat flux, allowing for manufacturing tolerances on fuel pellets and rods.
- $F_{\Delta H}^N$ Nuclear Enthalpy Rise Hot Channel Factor, is defined as the ratio of the integral of linear power along the rod with the highest integrated power to the average rod power.
- $F_{xy}(Z)$ Radial Peaking Factor, is defined as the ratio of peak power density to average power density in the horizontal plane at core elevation Z.

3/4.2.1 AXIAL FLUX DIFFERENCE (AFD)

The limits on AXIAL FLUX DIFFERENCE assure that the $F_Q(Z)$ upper bound envelope of 2.32 times the normalized axial peaking factor is not exceeded during either normal operation or in the event of xenon redistribution following power changes.

Target flux difference is determined at equilibrium xenon conditions with the parū length control rods withdrawn from the core. The full length rods may be positioned within the core in accordance with their respective insertion limits and should be inserted near their normal position for steady state operation at high power levels. The value of the target flux difference obtained under these conditions divided by the fraction of RATED THERMAL POWER is the target flux difference at RATED THERMAL POWER for the associated core burnup conditions. Target flux differences for other THERMAL POWER levels are obtained by multiplying the RATED THERMAL POWER value by the appropriate fractional THERMAL POWER level. The periodic updating of the target flux difference value is necessary to reflect core burnup considerations.

POWER DISTRIBUTION LIMITS

BASES

Although it is intended that the plant will be operated with the AXIAL FLUX DIFFERENCE within the +5% target band about the target flux difference, during rapid plant THERMAL POWER reductions, control rod motion will cause the AFD to deviate outside of the target band at reduced THERMAL POWER levels. This deviation will not affect the xenon redistribution sufficiently to change the envelope of peaking factors which may be reached on a subsequent return to RATED THERMAL POWER (with the AFD within the target band) provided the time duration of the deviation is limited. Accordingly, a 1 hour penalty deviation limit cumulative during the previous 24 hours is provided for operation outside of the target band but within the limits of Figure 3.2-1 while at THERMAL POWER levels between 50% & 90% of RATED THERMAL POWER. For THERMAL POWER levels between 15% & 50% of rated THERMAL POWER, deviations of the AFD outside of the target band are less significant. The penalty of 2 hours actual time reflects this reduced significance.

Provisions for monitoring the AFD on an automatic basis are derived from the plant process computer through the AFD Monitor Alarm. The computer determines the one minute average of each of the OPERABLE excore detector outputs and provides an alarm message immediately if the AFD for at least 3 of 4 or 2 of 3 OPERABLE excore channels are outside the target band & the THERMAL POWER is greater than 90% of RATED THERMAL POWER. During operation at THERMAL POWER levels between 50% & 90% & 15% & 50% RATED THERMAL POWER, the computer outputs an alarm message when the penalty deviation accumulates beyond the limits of 1 hour & 2 hours, respectively.

Figure B 3/4 2-1 shows a typical monthly target band near the beginning of core life.

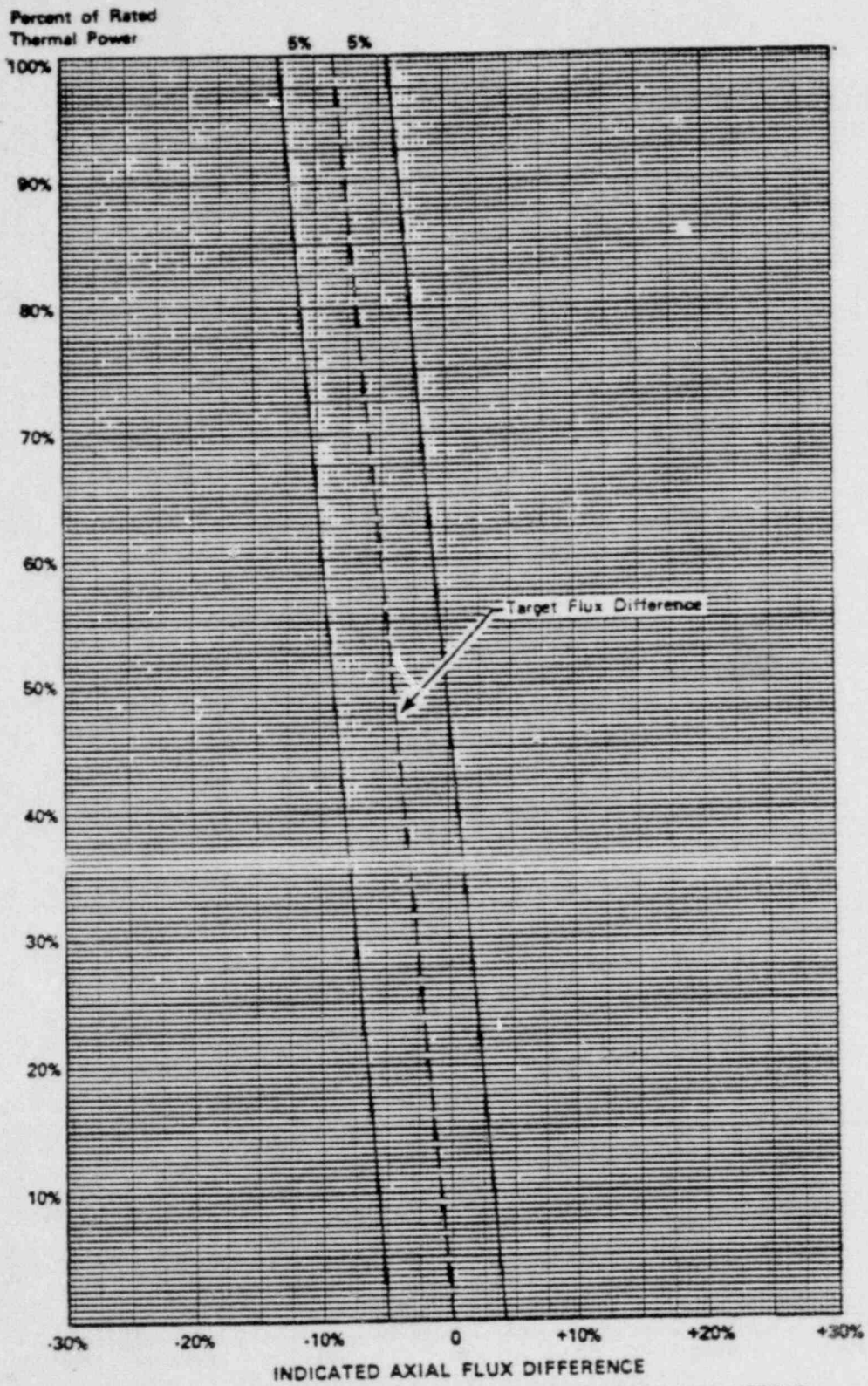


Figure B 3/4 2-1 TYPICAL INDICATED AXIAL FLUX DIFFERENCE VERSUS THERMAL POWER AT BOL

POWER DISTRIBUTION LIMITS

BASES

3/4.2.2 and 3/4.2.3 HEAT FLUX HOT CHANNEL FACTOR, RCS FLOWRATE, AND ENTHALPY RISE HOT CHANNEL FACTOR

The limits on heat flux hot channel factor, RCS flowrate, and nuclear enthalpy rise hot channel factor ensure that 1) the design limits on peak local power density and minimum DNBR are not exceeded and 2) in the event of a LOCA the peak fuel clad temperature will not exceed the 2200°F ECCS acceptance criteria limit.

Each of these is measurable but will normally only be determined periodically as specified in Specifications 4.2.2 and 4.2.3. This periodic surveillance is sufficient to insure that the limits are maintained provided:

- a. Control rods in a single group move together with no individual rod insertion differing by more than ± 12 steps from the group demand position.
- b. Control rod groups are sequenced with overlapping groups as described in Specification 3.1.3.5.
- c. The control rod insertion limits of Specifications 3.1.3.5 and 3.1.3.6 are maintained.
- d. The axial power distribution, expressed in terms of AXIAL FLUX DIFFERENCE, is maintained within the limits.

F_{AH}^N will be maintained within its limits provided conditions a. through d. above are maintained. As noted on Figures 3.2-3 and 3.2-4, RCS flowrate and F_{AH}^N may be "traded off" against one another (i.e., a low measured RCS flowrate is acceptable if the measured F_{AH}^N is also low) to ensure that the calculated DNBR will not be below the design DNBR value. This tradeoff is allowed up to a maximum F_{AH}^N of 1.49 (1+0.2(1-P)) which is consistent with the initial conditions assumed for the LOCA analysis. The relaxation of F_{AH}^N as a function of THERMAL POWER allows changes in the radial power shape for all permissible rod insertion limits.

When an F_0 measurement is taken, both experimental error and manufacturing tolerance must be allowed for. Five percent is the appropriate allowance for a full core map taken with the incore detector flux mapping system and 3% is the appropriate allowance for manufacturing tolerance. Application of these two penalties in a multiplication fashion is sufficient to provide a correction for the effect of rod bow on F_0 , which has been conservatively estimated as 5% in WCAP-8692, "Fuel Rod Bowing." The appropriate statistical combination of local power, manufacturing tolerance, and rod bow uncertainties results in a penalty on F_0 of 7.68%, whereas multiplying measured values of F_0 by 1.03 x 1.05 results in a penalty of 8.15%.

POWER DISTRIBUTION LIMITS

BASES

When RCS flowrate and $F_{\Delta H}^N$ are measured, no additional allowances are necessary prior to comparison with the limits of Figures 3.2-3 and 3.2-4. Measurement errors of 3.5% for RCS total flow rate and 4% for $F_{\Delta H}^N$ have been allowed for in determination of the design DNBR value.

The design DNBR values include a 20.2% margin for conservatism. The effect of rod bow on DNBR has been determined to be a 10.6% penalty. Therefore, the available margin more than offsets the effect of rod bow, and no penalty is required on DNBR or $F_{\Delta H}^N$.

3/4.2.4 QUADRANT POWER TILT RATIO

The quadrant power tilt ratio limit assures that the radial power distribution satisfies the design values used in the power capability analysis. Radial power distribution measurements are made during startup testing and periodically during power operation.

The limit of 1.02 at which corrective action is required provides DNB and linear heat generation rate protection with x-y plane power tilts. A limiting tilt of 1.025 can be tolerated before the margin for uncertainty in F_0 is depleted. The limit of 1.02 was selected to provide an allowance for the uncertainty associated with the indicated power tilt.

The 2-hour time allowance for operation with a tilt condition greater than 1.02 but less than 1.09 is provided to allow identification and correction of a dropped or misaligned rod. In the event such action does not correct the tilt, the margin for uncertainty on F_0 is reinstated by reducing the power by 3 percent for each percent of tilt in excess of 1.0.

POWER DISTRIBUTION LIMITS

BASES

3/4.2.5 DNB PARAMETERS

The limits on the DNB related parameters assure that each of the parameters are maintained within the normal steady state envelope of operation assumed in the transient and accident analyses. The limits are consistent with the initial FSAR assumptions and have been analytically demonstrated adequate to maintain a minimum DNBR of 1.73 throughout each analyzed transient.

The 12-hour periodic surveillance of these parameters thru instrument readout is sufficient to ensure that the parameters are restored within their limits following load changes and other expected transient operation. The 18-month periodic measurement of the RCS total flow rate is adequate to detect flow degradation and ensure correlation of the flow indication channels with measured flow such that the indicated percent flow will provide sufficient verification of flow rate on a 12-hour basis.

3/4.4 REACTOR COOLANT SYSTEM

BASES

3/4.4.1 REACTOR COOLANT LOOPS

The plant is designed to operate with all reactor coolant loops in operation, and maintain DNBR above 1.73 during all normal operations and anticipated transients. With one reactor coolant loop not in operation, THERMAL POWER is restricted to < 38 percent of RATED THERMAL POWER until the Overtemperature ΔT trip is reset. Either action ensures that the DNBR will be maintained above 1.73. A loss of flow in two loops will cause a reactor trip if operating above P-7 (10 percent of RATED THERMAL POWER) while a loss of flow in one loop will cause a reactor trip if operating above P-8 (38 percent of RATED THERMAL POWER).

A single reactor coolant loop provides sufficient heat removal capability for removing core decay heat while in HOT STANDBY; however, single failure considerations require placing a RHR loop into operation in the shutdown cooling mode if component repairs and/or corrective actions cannot be made within the allowable out-of-service time.

3/4.4.2 and 3/4.4.3 SAFETY VALVES

The pressurizer code safety valves operate to prevent the RCS from being pressurized above its Safety Limit of 2735 psig. Each safety valve is designed to relieve 420,000 lbs per hour of saturated steam at 110% of the valve's setpoint. The relief capacity of a single safety valve is adequate to relieve any overpressure condition which could occur during shutdown. In the event that no safety valves are OPERABLE, an operating RHR loop, connected to the RCS, provides overpressure relief capability and will prevent RCS overpressurization.

During operation, all pressurizer code safety valves must be OPERABLE to prevent the RCS from being pressurized above its safety limit of 2735 psig. The combined relief capacity of all of these valves is greater than the maximum surge rate resulting from a complete loss of load assuming no reactor trip until the first Reactor Protective System trip setpoint is reached (i.e., no credit is taken for a direct reactor trip on the loss of load) and also assuming no operation of the power-operated relief valves or steam dump valves.

Demonstration of the safety valves' lift settings will occur only during shutdown and will be performed in accordance with the provisions of Section XI of the ASME Boiler and Pressure Code, 1974 Edition.

REACTOR COOLANT SYSTEM

BASES

3/4.4.4 PRESSURIZER

A steam bubble in the pressurizer ensures that the RCS is not a hydraulically solid system and is capable of accommodating pressure surges during operation. The steam bubble also protects the pressurizer code safety valves and power operated relief valves against water relief. The power operated relief valves and steam bubble function to relieve RCS pressure during all design transients up to and including the design step load decrease with steam dump. Operation of the power operated relief valve minimizes the undesirable opening of the spring-loaded pressurizer code safety valves.

3/4.4.5 STEAM GENERATORS

One OPERABLE steam generator provides sufficient heat removal capability to remove decay heat after a reactor shutdown. The requirement for two OPERABLE steam generators, combined with other requirements of the Limiting Conditions for Operation ensures adequate decay heat removal capabilities for RCS temperatures greater than 350°F if one steam generator becomes inoperable due to single failure considerations. Below 350 ., decay heat is removed by the RHR system.

The Surveillance Requirements for inspection of the steam generator tubes ensure that the structural integrity of this portion of the RCS will be maintained. The program for inservice inspection of steam generator tubes is based on a modification of Regulatory Guide 1.83, Revision 1. Inservice inspection of steam generator tubing is essential in order to maintain surveillance of the conditions of the tubes in the event that there is evidence of mechanical damage or progressive degradation due to design, manufacturing errors, or inservice conditions that lead to corrosion. Inservice inspection of steam generator tubing also provides a means of characterizing the nature and cause of any tube degradation so that corrective measures can be taken.

The plant is expected to be operated in a manner such that the secondary coolant will be maintained within those parameter limits found to result in negligible corrosion of the steam generator tubes. If the secondary coolant chemistry is not maintained within these parameter limits, localized corrosion may likely result in stress corrosion cracking. The extent of cracking during plant operation would be limited by the limitation of steam generator tube leakage between the primary coolant