Attachment Ia

Marked-up Technical Specification Pages

Catawba

SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

BASES

3/4.3 INSTRUMENTATION

2.0 SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

2.1 SAFETY LIMITS

REACTOR CORE

[REPLACE WITH PARAGRAPHS FROM ATTACHED PAGE]

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 Figure 2.1-1a for four loop operation~~.

APPLICABILITY: MODES 1 and 2.

ACTION:

SHOWN IN FIGURE 2.1-1a (FOR UNIT 1) OR FIGURE 2.1-1b (FOR UNIT 2)

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, ~~and comply with the requirements of Specification 6.7.1~~. DETERMINE IF A SAFETY LIMIT HAS BEEN VIOLATED PRIOR TO STARTUP.

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, and comply with the requirements of Specification 6.7.1.

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, and comply with the requirements of Specification 6.7.1.

Insert for Technical Specification 2.1.1:

The maximum local fuel pin centerline temperature shall be less than 5080 - (6.5×10^{-3}) x (Burnup, MWD/MTU) °F. Operation within this limit is primarily assured by compliance with the overtemperature ΔT and overpower ΔT trip functions, including the axial imbalance limits.

The DNBR shall be maintained greater than the statistical limit for the BWCMV correlation. Operation within this limit is primarily assured by compliance with the overtemperature ΔT and overpower ΔT trip functions, including the axial imbalance limits.

REPLACE

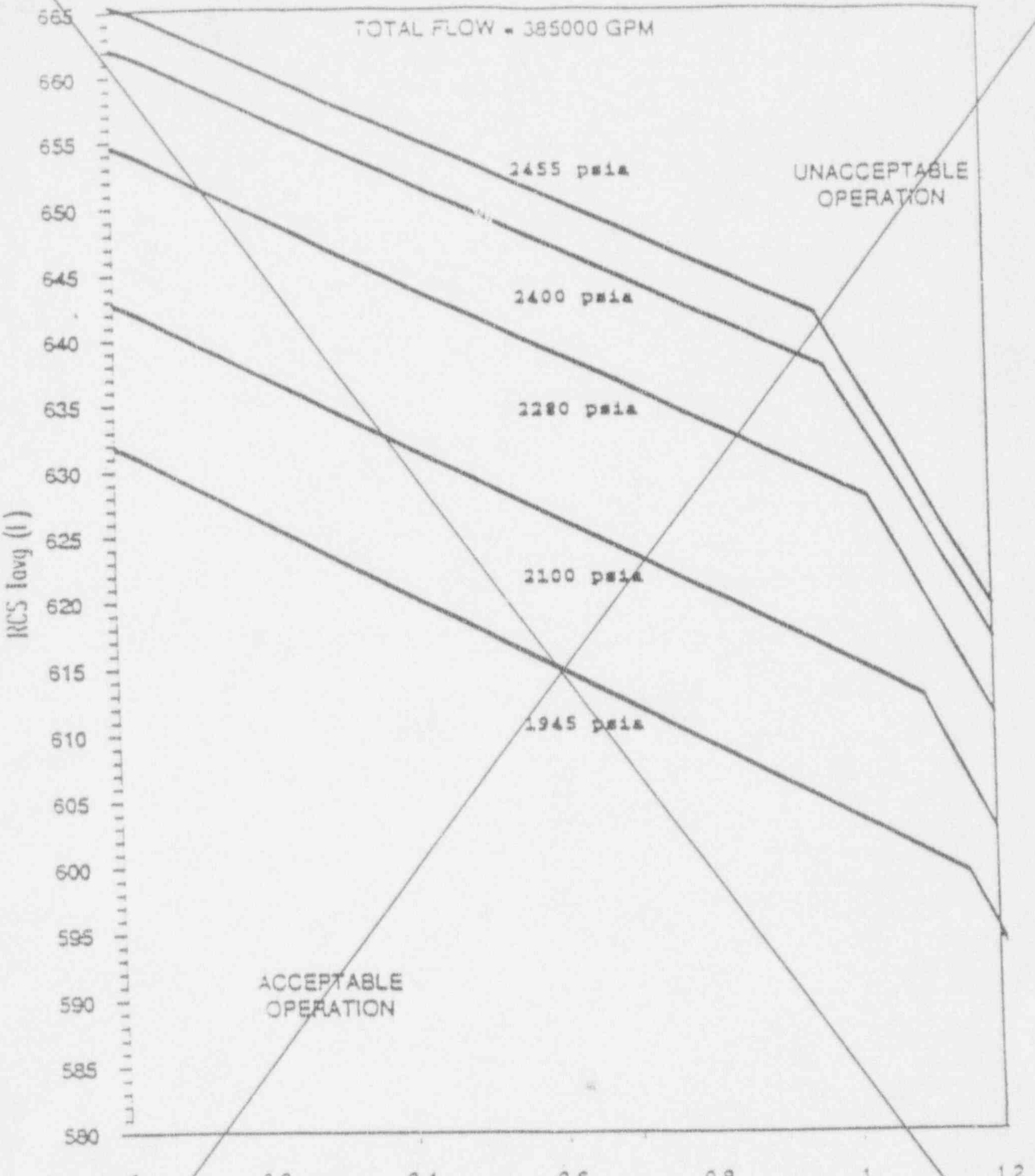


FIGURE 2.1-1
REACTOR CORE SAFETY LIMITS - FOUR LOOPS IN OPERATION

UNIT 1 ONLY

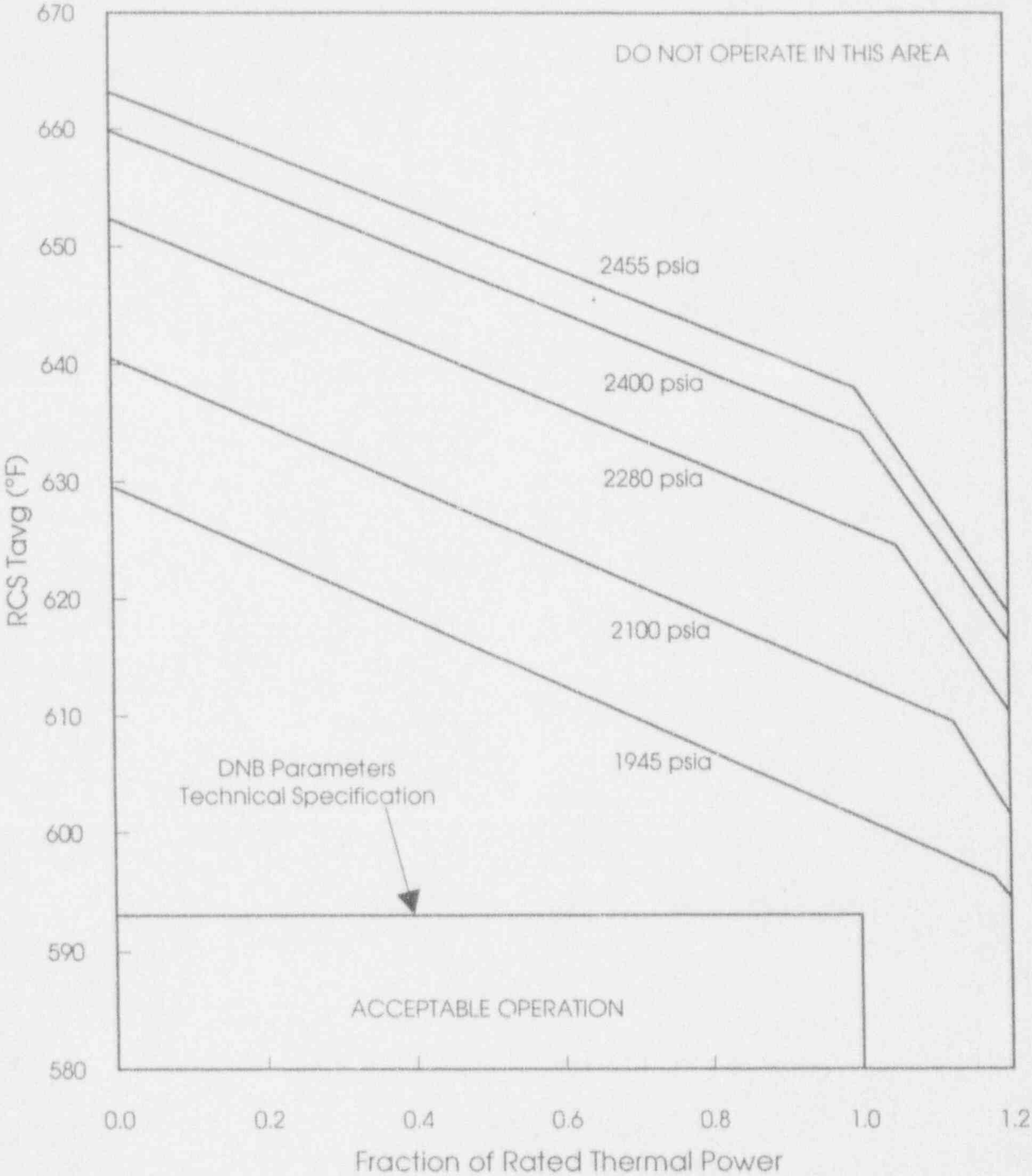


Figure 2.1-1a
REACTOR CORE PROTECTION LIMITS - FOUR LOOPS IN OPERATION

UNIT 2 ONLY

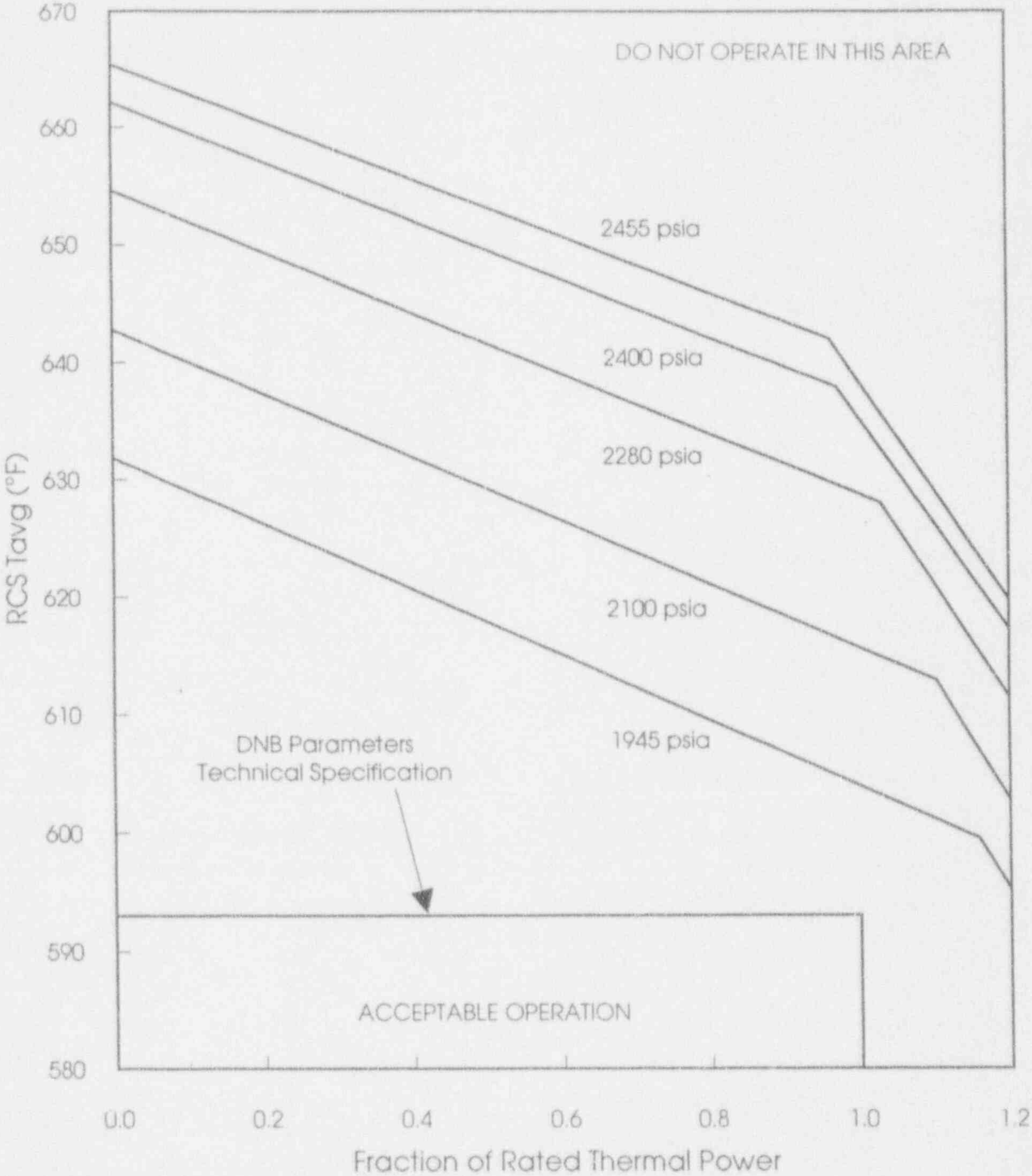


Figure 2.1-1b
REACTOR CORE PROTECTION LIMITS - FOUR LOOPS IN OPERATION

Unit 1 only

TABLE 2.2.-1

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUE
1. Manual Reactor Trip	N.A.	N.A.
2. Power Range, Neutron Flux		
a. High Setpoint	≤109% of RTP*	≤110.9% of RTP*
b. Low Setpoint	≤25% of RTP*	≤27.1% of RTP*
3. Power Range, Neutron Flux, High Positive Rate	≤5% of RTP* with a time constant ≥ 2 seconds	≤6.3% of RTP* with a time constant ≥ 2 seconds
4. Intermediate Range, Neutron Flux	≤25% of RTP*	≤31% of RTP*
5. Source Range, Neutron Flux	$\leq 10^5$ cps	$\leq 1.4 \times 10^5$ cps
6. Overtemperature ΔT	See Note 1	See Note 2
7. Overpower ΔT	See Note 3	See Note 4
8. Pressurizer Pressure-Low	≥1945 psig	≥1938 psig***
9. Pressurizer Pressure-High	≤2385 psig	≤2399 psig
10. Pressurizer Water Level-High	≤92% of instrument span	≤93.8% of instrument span
11. Reactor Coolant Flow-Low	≥90% of loop minimum measured flow**	≥88.9% of loop minimum measured flow**

*RTP = RATED THERMAL POWER

**Loop minimum measured flow = ~~96,250~~ 95 500 gpm

***Time constants utilized in the lead-lag controller for Pressurizer Pressure-Low are 2 seconds for lead and 1 second for lag. Channel calibration shall ensure that these time constants are adjusted to these values.

(Unit 2 only)

TABLE 2.2.-1

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUE
1. Manual Reactor Trip	N.A.	N.A.
2. Power Range, Neutron Flux		
a. High Setpoint	≤109% of RTP*	≤110.9% of RTP*
b. Low Setpoint	≤25% of RTP*	≤27.1% of RTP*
3. Power Range, Neutron Flux, High Positive Rate	≤5% of RTP* with a time constant ≥ 2 seconds	≤6.3% of RTP* with a time constant ≥ 2 seconds
4. Intermediate Range, Neutron Flux	≤25% of RTP*	≤31% of RTP*
5. Source Range, Neutron Flux	$\leq 10^5$ cps	$\leq 1.4 \times 10^5$ cps
6. Overtemperature ΔT	See Note 1	See Note 2
7. Overpower ΔT	See Note 3	See Note 4
8. Pressurizer Pressure-Low	≥1945 psig	≥1938 psig***
9. Pressurizer Pressure-High	≤2385 psig	≤2399 psig
10. Pressurizer Water Level-High	≤92% of instrument span	≤93.8% of instrument span
11. Reactor Coolant Flow-Low	≥90% of loop minimum measured flow**	≥88.9% of loop minimum measured flow**

*RTP = RATED THERMAL POWER
**Loop minimum measured flow = 96,250 gpm
***Time constants utilized in the lead-lag controller for Pressurizer Pressure-Low are 2 seconds for lead and 1 second for lag. Channel calibration shall ensure that these time constants are adjusted to these values.

Unit 1 only

TABLE 2.2-1 (Continued)
TABLE NOTATIONS

NOTE 1: OVERTEMPERATURE ΔT

$$\Delta T \frac{(1 + \tau_1 S)}{(1 + \tau_2 S)} \left(\frac{1}{1 + \tau_3 S}\right) \leq \Delta T_o \left\{K_1 - K_2 \frac{(1 + \tau_4 S)}{(1 + \tau_5 S)} \left[T \left(\frac{1}{1 + \tau_6 S}\right) - T'\right] + K_3 (P - P') - f_1 (\Delta I)\right\}$$

Where: ΔT = Measured ΔT by Loop Narrow Range RTDs;

$\frac{1 + \tau_1 S}{1 + \tau_2 S}$ = Lead-lag compensator on measured ΔT;

τ_1, τ_2 = Time constants utilized in lead-lag compensator for ΔT, τ_1 = 12 s, τ_2 = 3 s;

$\frac{1}{1 + \tau_3 S}$ = Lag compensator on measured ΔT;

τ_3 = Time constants utilized in the lag compensator for ΔT, τ_3 = 0;

ΔT_o = Indicated ΔT at RATED THERMAL POWER;

K_1 = ~~1.1953~~ 1.1954

K_2 = ~~0.03163~~/°F 0.03371

$\frac{1 + \tau_4 S}{1 + \tau_5 S}$ = The function generated by the lead-lag compensator for T_{avg} dynamic compensation;

τ_4, τ_5 = Time constants utilized in the lead-lag compensator for T_{avg}, τ_4 = 22 s, τ_5 = 4 s;

T = Average temperature, °F;

$\frac{1}{1 + \tau_6 S}$ = Lag compensator on measured T_{avg};

τ_6 = Time constant utilized in the measured T_{avg} lag compensator, τ_6 = 0;

Unit 2 only

TABLE 2.2-1 (Continued)
TABLE NOTATIONS

NOTE 1: OVERTEMPERATURE ΔT

$$\Delta T \frac{(1 + \tau_1 S)}{(1 + \tau_2 S)} \left(\frac{1}{1 + \tau_3 S}\right) \leq \Delta T_o \left(K_1 - K_2 \frac{(1 + \tau_4 S)}{(1 + \tau_5 S)} \left[T \left(\frac{1}{1 + \tau_6 S}\right) - T'\right] + K_3 (P - P') - f_1 (\Delta I)\right)$$

Where: ΔT = Measured ΔT by Loop Narrow Range RTDs;

$\frac{1 + \tau_1 S}{1 + \tau_2 S}$ = Lead-lag compensator on measured ΔT;

τ_1, τ_2 = Time constants utilized in lead-lag compensator for ΔT, τ_1 = 12 s, τ_2 = 3 s;

$\frac{1}{1 + \tau_3 S}$ = Lag compensator on measured ΔT;

τ_3 = Time constants utilized in the lag compensator for ΔT, τ_3 = 0;

ΔT_o = Indicated ΔT at RATED THERMAL POWER;

K_1 = 1.1953

K_2 = 0.03163/°F

$\frac{1 + \tau_4 S}{1 + \tau_5 S}$ = The function generated by the lead-lag compensator for T_{avg} dynamic compensation;

τ_4, τ_5 = Time constants utilized in the lead-lag compensator for T_{avg}, τ_4 = 22 s, τ_5 = 4 s;

T = Average temperature, °F;

$\frac{1}{1 + \tau_6 S}$ = Lag compensator on measured T_{avg};

τ_6 = Time constant utilized in the measured T_{avg} lag compensator, τ_6 = 0;

Unit 1 only

1.

NOTE 1: (Continued)

$T' \leq$ 590.8°F (Nominal T_{avg} allowed by Safety Analysis);

K_3 = ~~0.001414~~; 0.001529

P = Pressurizer pressure, psig;

P' = 2235 psig (Nominal RCS operating pressure);

S = Laplace transform operator, s^{-1};

and $f_1(\Delta I)$ is a function of the indicated difference between top and bottom detectors of the power-range neutron 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 ~~-39.9%~~ -42.0 and ~~+3.0%~~ +8.0,

$f_1(\Delta I)$ = 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 ΔI that the magnitude of q_t - q_b is more negative than ~~-39.9%~~ -42.0, the ΔT Trip Setpoint shall be automatically reduced by ~~3.910%~~ 3.672 of ΔT_o; and

(iii) For each percent ΔI that the magnitude of q_t - q_b is more positive than ~~+3.0%~~ +8.0, the ΔT Trip Setpoint shall be automatically reduced by ~~2.316%~~ 1.640 of ΔT_o.

NOTE 2: The channel's maximum Trip Setpoint shall not exceed its computed Trip Setpoint by more than ~~3.0%.~~ 4.5% Rated Thermal Power

Unit 2 only

1.

TABLE 2.2-1 (Continued)
TABLE NOTATIONS (Continued)

NOTE 1: (Continued)

$T' \leq 590.8°F$ (Nominal T_{avg} allowed by Safety Analysis);

K_3 = 0.001414;

P = Pressurizer pressure, psig;

P' = 2235 psig (Nominal RCS operating pressure);

S = Laplace transform operator, s^{-1};

and $f_1(\Delta I)$ is a function of the indicated difference between top and bottom detectors of the power-range neutron 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 -39.9% and +3.0%,

$f_1(\Delta I) = 0$, 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 ΔI that the magnitude of $q_t - q_b$ is more negative than -39.9%, the ΔT Trip Setpoint shall be automatically reduced by 3.910% of ΔT_o; and

(iii) For each percent ΔI that the magnitude of $q_t - q_b$ is more positive than +3.0%, the ΔT Trip Setpoint shall be automatically reduced by 2.316% of ΔT_o.

NOTE 2: The channel's maximum Trip Setpoint shall not exceed its computed Trip Setpoint by more than ~~3.0%~~. 4.5 % Rated Thermal Power

Unit 1 only

1.

TABLE 2.2-1 (Continued)
TABLE NOTATIONS (Continued)

NOTE 3: OVERPOWER ΔT

$$\Delta T \frac{(1 + \tau_1 S)}{(1 + \tau_2 S)} \left(\frac{1}{1 + \tau_3 S}\right) \leq \Delta T_o \{K_4 - K_5 \left(\frac{\tau_7 S}{1 + \tau_7 S}\right)\left(\frac{1}{1 + \tau_6 S}\right) T - K_6 [T \left(\frac{1}{1 + \tau_6 S}\right) - T''] - f_2(\Delta I)\}$$

Where: ΔT = As defined in Note 1,

$\frac{1 + \tau_1 S)}{1 + \tau_2 S}$ = As defined in Note 1,

τ_1, τ_2 = As defined in Note 1,

$\frac{1}{1 + \tau_3 S}$ = As defined in Note 1,

τ_3 = As defined in Note 1,

ΔT_o = As defined in Note 1,

K_4 = ~~1.0819~~ 1.0855

K_5 = 0.02/°F for increasing average temperature and 0 for decreasing average temperature,

$\frac{\tau_7 S}{1 + \tau_7 S}$ = The function generated by the rate-lag controller for T_{avg} dynamic compensation,

τ_7 = Time constant utilized in the rate-lag controller for T_{avg}, τ_7 = 10 s,

$\frac{1}{1 + \tau_6 S}$ = As defined in Note 1,

τ_6 = As defined in Note 1,

Unit 2 only

TABLE 2.2-1 (Continued)
TABLE NOTATIONS (Continued)

NOTE 3: OVERPOWER ΔT

$$\Delta T \frac{(1 + \tau_1 S)}{(1 + \tau_2 S)} \left(\frac{1}{1 + \tau_3 S}\right) \leq \Delta T_o \left\{K_4 - K_5 \left(\frac{\tau_7 S}{1 + \tau_7 S}\right) \left(\frac{1}{1 + \tau_6 S}\right) T - K_6 \left[T \left(\frac{1}{1 + \tau_6 S}\right) - T''\right] - f_2(\Delta I)\right\}$$

Where: ΔT = As defined in Note 1,

$\frac{1 + \tau_1 S)}{1 + \tau_2 S}$ = As defined in Note 1,

τ_1, τ_2 = As defined in Note 1,

$\frac{1}{1 + \tau_3 S}$ = As defined in Note 1,

τ_3 = As defined in Note 1,

ΔT_o = As defined in Note 1,

K_4 = 1.0819

K_6 = 0.02/°F for increasing average temperature and 0 for decreasing average temperature,

$\frac{\tau_7 S}{1 + \tau_7 S}$ = The function generated by the rate-lag controller for T_{avg} dynamic compensation,

τ_7 = Time constant utilized in the rate-lag controller for T_{avg}, τ_7 = 10 s,

$\frac{1}{1 + \tau_6 S}$ = As defined in Note 1,

τ_6 = As defined in Note 1,

Unit 1 only

TABLE 2.2-1 (Continued)

TABLE NOTATIONS (Continued)

NOTE 3: (Continued)

K_6 = ~~0.001291~~ 0.001262/°F for T > 590.8°F and K_6 = 0 for T ≤ 590.8°F,

T = As defined in Note 1,

T" = Indicated T_{avg} at RATED THERMAL POWER (Calibration temperature for ΔT instrumentation, ≤ 590.8°F),

S = As defined in Note 1,

and f_2 (ΔI) is a function of the indicated differences between top and bottom detectors of the power-range neutron 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 -35% and +35% ΔI; f_2 (ΔI) = 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 ΔI that the magnitude of $q_t - q_b$ is more negative than -35% ΔI, the ΔT Trip Setpoint shall be automatically reduced by 7.0% of ΔT_o; and

(iii) for each percent ΔI that magnitude of $q_t - q_b$ is more positive than +35% ΔI, the ΔT Trip Setpoint shall be automatically reduced by 7.0% of ΔT_o.

NOTE 4: The channel's maximum Trip Setpoint shall not exceed its computed Trip Setpoint by more than ~~2.8%.~~ 3.0 % Rated Thermal Power

Unit 2 only

TABLE 2.2-1 (Continued)
TABLE NOTATIONS (Continued)

NOTE 3: (Continued)

K_6 = 0.001291/°F for T > 590.8°F and K_6 = 0 for T ≤ 590.8°F,

T = As defined in Note 1,

T" = Indicated T_{avg} at RATED THERMAL POWER (Calibration temperature for ΔT instrumentation, ≤ 590.8°F),

S = As defined in Note 1,

and f_2 (ΔI) is a function of the indicated differences between top and bottom detectors of the power-range neutron 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 -35% and +35% ΔI; f_2 (ΔI) = 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 ΔI that the magnitude of q_t - q_b is more negative than -35% ΔI, the ΔT Trip Setpoint shall be automatically reduced by 7.0% of ΔT_o; and

(iii) for each percent ΔI that magnitude of q_t - q_b is more positive than +35% ΔI, the ΔT Trip Setpoint shall be automatically reduced by 7.0% of ΔT_o.

NOTE 4: The channel's maximum Trip Setpoint shall not exceed its computed Trip Setpoint by more than ~~2.8%.~~
3.3% of Rated Thermal Power

Unit 1 only

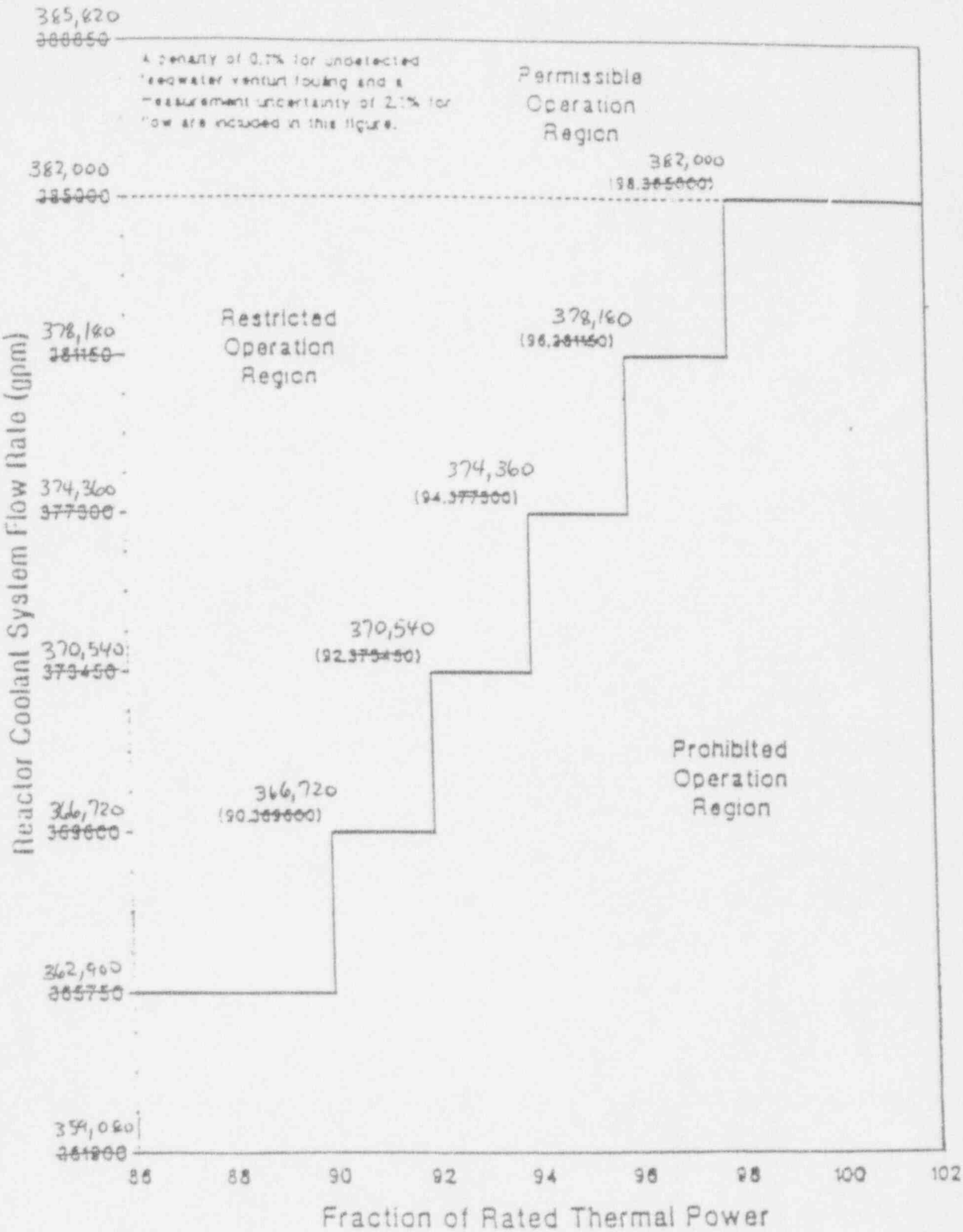


Figure 3.2-1 Reactor Coolant System Total Flow Rate Versus Rated Thermal Power - Four Loops in Operation

Unit 2 only

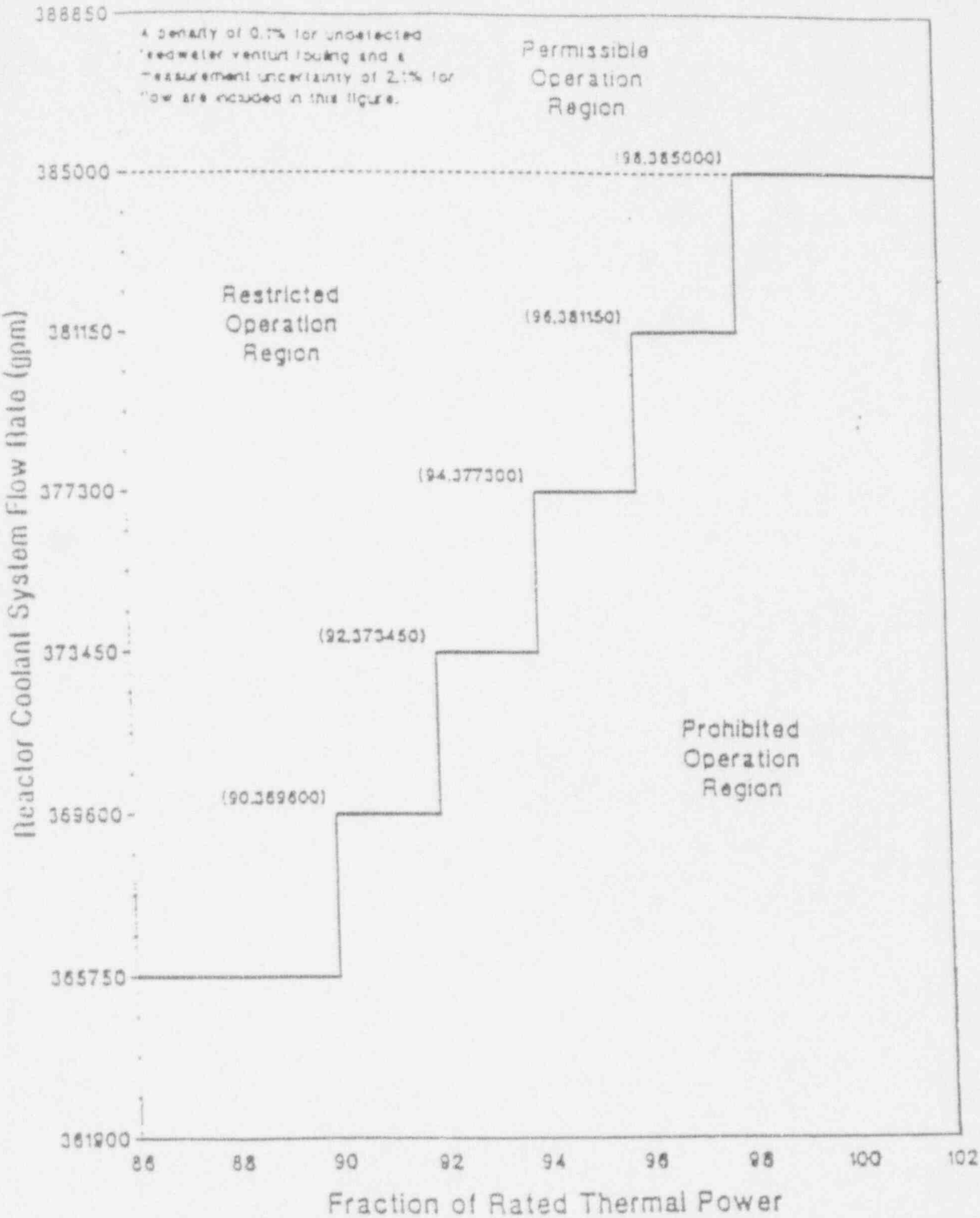


Figure 3.2-1 Reactor Coolant System Total Flow Rate Versus Rated Thermal Power - Four Loops in Operation

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 , Flow, and Pressure have been related to DNB through the BWCMV correlation. The BWCMV DNB correlation has been developed to predict the DNB flux and the location of DNB for axially uniform and nonuniform heat flux distributions. The local DNB heat flux ratio, (DNBR), is defined as the ratio of the heat flux that would cause DNB at a particular core location to the local heat flux, and is indicative of the margin to DNB.

The DNB design basis is as follows: there must be at least a 95% probability that the minimum DNBR of the limiting rod during Condition I and II events is greater than or equal to the DNBR limit of the DNB correlation being used (the BWCMV correlation in this application). The correlation DNBR limit is established based on the entire applicable experimental data set such that there is a 95% probability with 95% 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, and the BWCMV DNB correlation are considered statistically such that there is at least a 95% confidence that the minimum DNBR for the limited rod is greater than or equal to the DNBR limit. The uncertainties in the above parameters are used to determine the plant DNBR uncertainty. This DNBR uncertainty is used to establish a design DNBR value which must be met in plant safety analyses using values of input parameters without uncertainties.

The curves of Figure 2.1-1a for Unit 1 and 2.1-1b for Unit 2, ~~2.1-1 show the loci of points of THERMAL POWER, Reactor Coolant System pressure and average temperature below 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.~~ are more restrictive than the actual safety limit curves.

The OTΔT and OPΔT trip functions are

~~These curves are~~ based on a nuclear enthalpy rise hot channel factor, $F^N_{\Delta H}$, of 1.50 and a reference cosine with a peak of 1.55. An allowance is included for an increase in $F^N_{\Delta H}$ at reduced power based on the expression:

$F^N_{\Delta H} = 1.50 [1 + 1/RRH (1-P)]$

Where P is the fraction of RATED THERMAL POWER.
RRH is given in the COLR.

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_1 (ΔI) 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 Setpoints to provide protection consistent with core Safety Limits.

Insert paragraph from attached page

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 vessel, pressurizer, and the Reactor Coolant System piping, valves, and fittings are designed to Section III of the ASME Code for Nuclear Power Plants which permits a maximum transient pressure of 110% (2735 psig) of 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 125% (3110 psig) of design pressure, to demonstrate integrity prior to initial operation.

Insert for Technical Specification 2.1.1 Bases

Since pin power peaking is not directly measurable, fuel melt limited power peaks are separately correlated to measured reactor power and imbalance. When the combination of reactor power and axial power imbalance is not within tolerance, the OPDT trip function will provide the necessary fuel pin centerline temperature protection.

Attachment Ib

Marked-up Technical Specification Pages

McGuire

INDEX

SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

BASES

2.0 SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

2.1 SAFETY LIMITS

[REPLACE W/ PARAGRAPH ON ATTACHED PAGE]

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 four and three loop operation, respectively.~~

APPLICABILITY: MODES 1 and 2

ACTION:

SHOWN IN FIGURE 2.1-1

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, ~~and comply with the requirements of Specification 6.7.1.~~ DETERMINE IF A SAFETY LIMIT HAS BEEN VIOLATED PRIOR TO STARTUP.

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, and comply with the requirements of Specification 6.7.1.

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, and comply with the requirements of Specification 6.7.1.

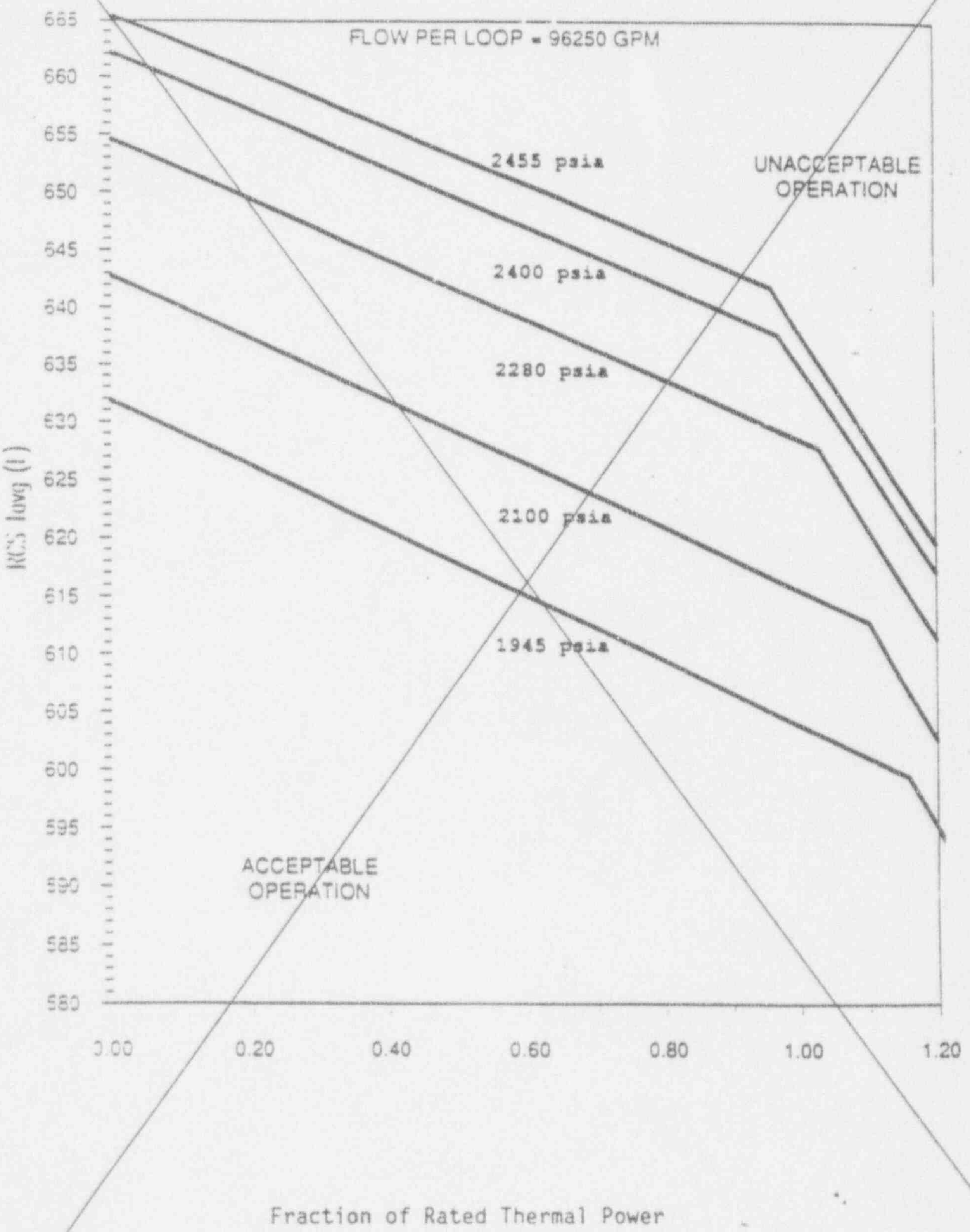
Insert for Technical Specification 2.1.1:

The maximum local fuel pin centerline temperature shall be less than $5080 - (6.5 \times 10^{-3}) \times$ (Burnup, MWD/MTU) °F. Operation within this limit is primarily assured by compliance with the overtemperature ΔT and overpower ΔT trip functions, including the axial imbalance limits.

The DNBR shall be maintained greater than the statistical limit for the BWCMV correlation. Operation within this limit is primarily assured by compliance with the overtemperature ΔT and overpower ΔT trip functions, including the axial imbalance limits.

REPLACE

Figure 2.1-1 Reactor Core Safety Limits - Four Loops in Operation



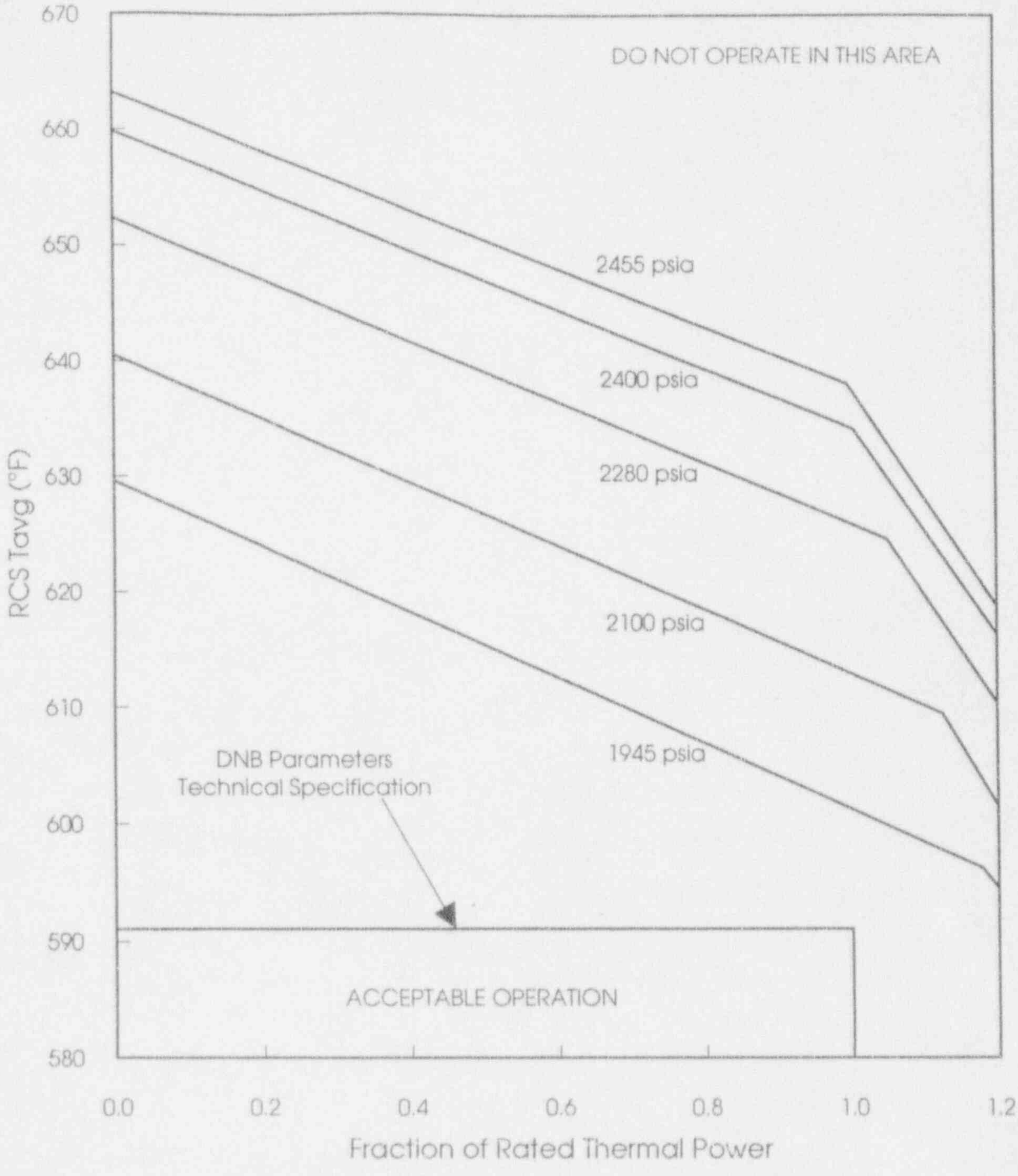


Figure 2.1-1
REACTOR CORE PROTECTION LIMITS - FOUR LOOPS IN OPERATION

DELETE

Figure 2.1-2 left blank pending NRC approval of three loop operation

TABLE 2.2-1

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
1. Manual Reactor Trip	N.A.	N.A.
2. Power Range, Neutron Flux	Low Setpoint - $\leq$ 25% of RATED THERMAL POWER	Low Setpoint - $\leq$ 26% of RATED THERMAL POWER
	High Setpoint - $\leq$ 109% 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 $\geq$ 2 seconds	$\leq$ 5.5% of RATED THERMAL POWER with a time constant $\geq$ 2 seconds
4. Intermediate Range, Neutron Flux	$\leq$ 25% of RATED THERMAL POWER	$\leq$ 30% of RATED THERMAL POWER
5. Source Range, Neutron Flux	$\leq 10^5$ counts per second	$\leq 1.3 \times 10^5$ counts per second
6. Overtemperature ΔT	See Note 1	See Note 3
7. Overpower ΔT	See Note 2	See Note 4
8. Pressurizer Pressure--Low	$\geq$ 1945 psig	$\geq$ 1935 psig
9. Pressurizer Pressure--High	$\leq$ 2385 psig	$\leq$ 2395 psig
10. Pressurizer Water Level--High	$\leq$ 92% of instrument span	$\leq$ 93% of instrument span
11. Low Reactor Coolant Flow	$\geq$ 90% of minimum measured flow per loop*	$\geq$ 88.8% of minimum measured flow per loop*

*Minimum measured flow is ~~96,250~~ 95 500 gpm per loop.

TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

NOTATION

NOTE 1: OVERTEMPERATURE ΔT

$$(\Delta T/\Delta T_o)\ \left(\frac{1 + \tau_1 S}{1 + \tau_2 S}\right)\left(\frac{1}{1 + \tau_3 S}\right) \leq K_1 - K_2 \left(\frac{1 + \tau_4 S}{1 + \tau_5 S}\right)\left[T\left(\frac{1}{1 + \tau_6 S}\right) - T'\right] + K_3(P - P') - f_1(\Delta I)$$

Where: ΔT = Measured ΔT by Loop Narrow Range RTD

ΔT_o = Indicated ΔT at RATED THERMAL POWER,

$\frac{1 + \tau_1 S}{1 + \tau_2 S}$ = Lead-lag compensator on measured ΔT,

τ_1, τ_2 = Time constants utilized in the lead-lag controller for ΔT, $\tau_1 \geq 8$ sec., $\tau_2 \leq 3$ sec.,

$\frac{1}{1 + \tau_3}$ = Lag compensator on measured ΔT,

τ_3 = Time constants utilized in the lag compensator for ΔT, $\tau_3 \leq$ 2 sec.*

K_1 ≤ ~~1.1958,~~ 1.1988

K_2 = ~~0.03143~~ 0.03354

$\frac{1 + \tau_4 S}{1 + \tau_5 S}$ = The function generated by the lead-lag controller for T_{avg} dynamic compensation,

τ_4, τ_5 = Time constants utilized in the lead-lag controller for T_{avg}, $\tau_4 \geq 28$ sec, $\tau_5 \leq 4$ sec.,

T = Average temperature, °F,

$\frac{1}{1 + \tau_6 S}$ = Lag compensator on measured T_{avg},

TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

NOTATION (Continued)

NOTE 1: (Continued)

τ_6 = Time constant utilized in the measured T_{avg} lag compensator, $\tau_6 \leq 2$ sec

T' = $\leq$ 588.2°F Reference T_{avg} at RATED THERMAL POWER,

K_3 = ~~0.001405~~, 0.001522

P = Pressurizer pressure, psig,

P' = 2235 psig (Nominal RCS operating pressure),

S = Laplace transform operator, sec^{-1},

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 ~~-39%~~ -44.0 and ~~+7.0%~~ +12.0 ΔI; $f_1(\Delta I) = 0$, 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 imbalance that the magnitude of $q_t - q_b$ is more negative than ~~-39%~~ -44.0 ΔI, the ΔT Trip Setpoint shall be automatically reduced by ~~6.153%~~ 3.436 of ΔT_o, and

(iii) for each percent imbalance that the magnitude of $q_t - q_b$ is more positive than ~~+7.0%~~ +12.0 ΔI, the ΔT Trip Setpoint shall be automatically reduced by ~~1.511%~~ 1.619 of ΔT_o

TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

NOTATION (Continued)

NOTE 2: OVERPOWER ΔT

$$(\Delta T/\Delta T_0) \left(\frac{1 + \tau_1 S}{1 + \tau_2 S}\right) \left(\frac{1}{1 + \tau_3 S}\right) \leq K_4 - K_5 \left(\frac{\tau_7 S}{1 + \tau_7 S}\right) \left(\frac{1}{1 + \tau_6 S}\right) T - K_6 \left[T\left(\frac{1}{1 + \tau_6 S}\right) - T''\right] - f_2(\Delta I)$$

Where: ΔT = As defined in Note 1,

ΔT_0 = As defined in Note 1,

$\frac{1 + \tau_1 S}{1 + \tau_2 S}$ = As defined in Note 1

τ_1, τ_2 = As defined in Note 1

$\frac{1}{1 + \tau_3 S}$ = As defined in Note 1,

K_4 $\leq$ ~~1.0809,~~ 1.0851

K_5 = 0.02/°F for increasing average temperature and 0 for decreasing average temperature,

$\frac{\tau_7 S}{1 + \tau_7 S}$ = The function generated by the rate-lag controller for T_{avg} dynamic compensation,

τ_7 = Time constant utilized in the rate-lag controller for T_{avg}, $\tau_7 \geq 5$ sec,

$\frac{1}{1 + \tau_6 S}$ = As defined in Note 1,

τ_6 = As defined in Note 1,

K_6 = ~~0.001239~~/°F for T > T" and K_6 = 0 for T $\leq$ T",
0.001207

TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

NOTATION (Continued)

T = As defined in Note 1,

T" = $\leq$ 588.2°F Reference T_{avg} at RATED THERMAL POWER,

S = As defined in Note 1, and

$f_2(\Delta I)$ 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 -35% and +35% ΔI; $f_2(\Delta I) = 0$, 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 imbalance that the magnitude of $q_t - q_b$ is more negative than -35% ΔI, the ΔT Trip Setpoint shall be automatically reduced by 7.0% of ΔT_o; and

(iii) for each percent imbalance that the magnitude of $q_t - q_b$ is more positive than +35% ΔI, the ΔT Trip Setpoint shall be automatically reduced by 7.0% of ΔT_o.

Note 3: The channel's maximum Trip Setpoint shall not exceed its computed Trip Setpoint by more than ~~3.6~~ 4.4% of Rated Thermal Power.

Note 4: The channel's maximum Trip Setpoint shall not exceed its computed Trip Setpoint by more than ~~4.2~~ 3.0% of Rated Thermal Power.

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 nonuniform 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 as follows: there must be at least a 95% probability that the minimum DNBR of the limiting rod during Condition I and II events is greater than or equal to the DNBR limit of the DNB correlation being used (the BWCMV correlation in this application). The correlation DNBR set such that there is a 95% probability with 95% 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, fuel fabrication parameters, and the CHF correlation are considered statistically such that there is at least a 95% confidence that the minimum DNBR for the limiting rod is greater than or equal to the DNBR limit. The combined DNBR uncertainty is used to establish a design DNBR value which must be met in plant safety analyses using values of input parameters without uncertainties.

The curves of Figure 2.1-1 are more restrictive than the actual safety limits. ~~show the loci of points of THERMAL POWER, Reactor Coolant System pressure, and average temperature below 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.~~

The ~~curves~~ OTΔT and OPΔT trip functions are based on a nuclear enthalpy rise hot channel factor, $F^N_{\Delta H}$, of 1.50 and a reference cosine axial power shape with a peak of 1.55. An allowance is included for an increase in $F^N_{\Delta H}$ at reduced power based on the expression:

$$F^N_{\Delta H} = 1.50\ [1 + (1/RRH)\ (1-P)]$$

Where P is the fraction of RATED THERMAL POWER, and RRH is given in the COLR.

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_1 (ΔI) function of the Overtemperature trip. When the axial power imbalance is not within the tolerance, the axial power imbalance effect on the Overtemperature ΔT trips will reduce the setpoints to provide protection consistent with core safety limits.

Insert paragraph from attached page

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.

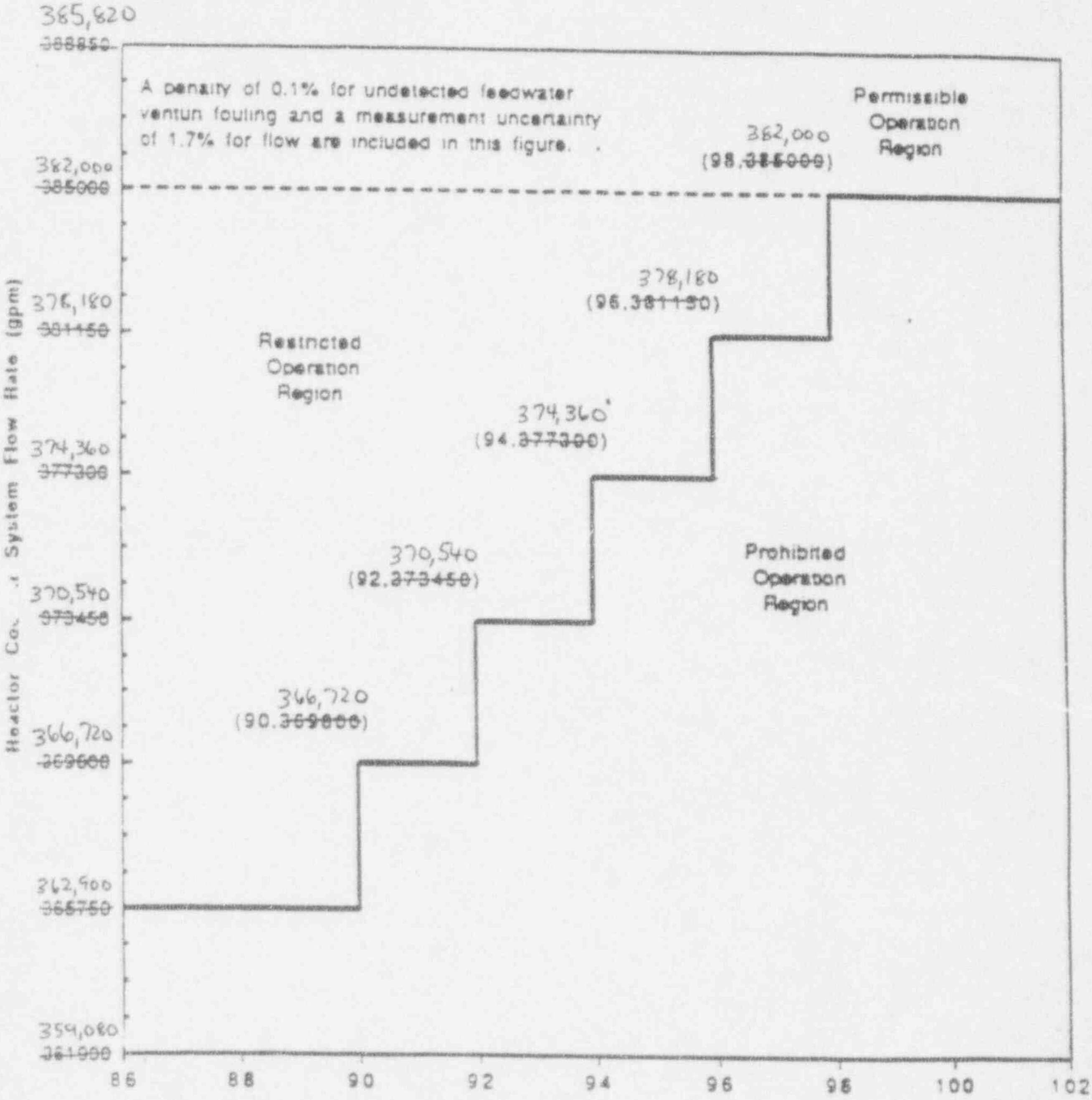
The reactor vessel and pressurizer are designed to Section III of the ASME Code for Nuclear Power Plants which permits a maximum transient pressure of 110% (2735 psig) of design pressure. The Safety Limit of 2735 psig is therefore consistent with the design criteria and associated code requirements.

The entire Reactor Coolant System is hydrotested at 3107 psig, 125% of design pressure, to demonstrate integrity prior to initial operation.

Insert for Technical Specification 2.1.1 Bases

Since pin power peaking is not directly measurable, fuel melt limited power peaks are separately correlated to measured reactor power and imbalance. When the combination of reactor power and axial power imbalance is not within tolerance, the OPDT trip function will provide the necessary fuel pin centerline temperature protection.

Figure 3.2 - 1. Reactor Coolant System Total Flow Rate Versus Rated Thermal Power - Four Loops in Operation



Fraction of Rated Thermal Power

Attachment II

Justification and Safety Analysis

Over time, degraded steam generator tubes have been plugged or sleeved, resulting in a reduction of reactor coolant system flow. In addition to this, the hot leg streaming phenomenon affects the accurate measurement of flow. As a result of these effects, it will become difficult to ensure meeting the minimum flow requirement (Table 2.2-1 Item 12, as annotated) required by Technical Specifications to maintain 100% power operation. The proposed reduction in minimum measured flow is applicable to McGuire Units 1 and 2, and to Catawba Unit 1.

To alleviate this concern, analyses have been performed to justify reduction in the minimum RCS flow to 382,000 gpm. These analyses show that the reduced flow rate will not have a significant impact on any accident analyses presented in Chapters 3, 4, 6, or 15 of the Final Safety Analysis Report (FSAR).

The overtemperature delta T (OTΔT) and overpower delta T (OPΔT) setpoint equation constants have been revised to support the reduction in minimum measured flow. The methodology used to generate the constants is described in the April 26, 1993 letter from T. C. McMeekin, Duke Power Company, to USNRC Document Control Desk, Supplement to Technical Specification Amendment Relocation of Cycle-Specific Limits to the Core Operating Limits Report. The proposed revision to the OTΔT and OPΔT constants is applicable to McGuire Units 1 and 2, and to Catawba Unit 1. The change is not applicable to Catawba Unit 2, because the steam generators in Unit 2 have not degraded and have not required tube plugging or sleeving to the extent of the other three units. This is consistent with Duke's current plans to replace the steam generators in both McGuire units, and Unit 1 only at Catawba. The higher minimum flow in Unit 2 is being retained to provide increased flexibility in fuel cycle design work.

The changes to the OPΔT setpoints for McGuire Units 1 and 2 and Catawba Unit 1 also necessitated recalculation of the Technical Specification allowable values of the trip functions. The revised OPΔT allowable values are more restrictive than the existing values. In the course of these calculations, a minor error was discovered that affected the existing allowable values for all four units. This resulted in a recalculation of the allowable value for Catawba Unit 2, as well as the three units affected by the flow reduction. Since the setpoint is, by administrative controls, reset whenever it is found to be different from the correct value by about 1%, it is unlikely that past operability will be a concern. Investigation into the situation is continuing, and a Licensee Event Report will be submitted, if appropriate. The revised McGuire OTΔT allowable value is less restrictive than the existing value and the Catawba allowable value is unchanged by the reduction in flow. Also, to

improve clarity, the allowable values of OPΔT and OTΔT are now expressed in units of % Rated Thermal Power.

DNBR and centerline fuel temperature (CFT) limits have been added to Technical Specification 2.1.1 as Limiting Conditions for Operation (LCO), replacing a reference to Figure 2.1-1, Reactor Core Safety Limit - Four Loops in Operation. This is considered to more accurately reflect the requirements of 10 CFR 50.36. A curve was added to the figure to show acceptable operation as determined by DNBR limits; this curve is more restrictive than the existing figure. This change is applicable to both units at each of the two stations. Figure 2.1-1 was redrawn to reflect the reduced flow rate for McGuire Units 1 and 2, and Catawba Unit 1. The change to DNBR and CFT in the LCO of Specification 2.1.1 is consistent with Babcock and Wilcox Improved Standard Technical Specification presented in NUREG-1430.

Effect of Reduced Flow on FSAR Analyses

LOCA Blowdown Forces, FSAR Chapter 3

The primary factors which affect the blowdown forces resulting from a LOCA are RCS pressure, vessel inlet and outlet fluid temperatures, and to a smaller degree, the loop and vessel flowrates. The LOCA analyses have been performed with a flow which corresponds to a minimum measured flow (MMF) less than 382000 gpm, and therefore a reduction in MMF to 382000 gpm will not affect the assumptions in the blowdown forces analysis.

Thermal Hydraulic Design, FSAR Section 4.4

The thermal hydraulic design for the McGuire and Catawba units was analyzed with the reduction in RCS minimum measured flow (MMF) to 382,000 gpm. The reduced flow rate resulted in a slight reduction of the margin in the core DNB limits. Technical Specification Figure 3.2-1, Reactor Coolant System Total Flow Rate Versus Rated Thermal Power - Four Loops In Operation, was revised to reflect the lower allowable flow rate. The Axial Flux Difference limits, Technical Specification Section 3.2.1, are unchanged and all of the current thermal hydraulic design criteria are satisfied at the reduced flow conditions.

As previously noted, revised core thermal limits were generated to reflect the reduced minimum measured RCS flow of 382,000 gpm. Based on these new protection limits, the overtemperature delta T (OTΔT) setpoint equation constants (Note 1 of Table 2.2-1), and the overpower delta T (OPΔT) setpoint equation constants (Notes 2 and 3 of Table 2.2-1 for McGuire and Catawba, respectively) were revised to reflect the necessary changes. The impact of the reduced flow on the coefficients was partially offset by a

reduction in the margin assumed in the calculation of the coefficients.

Mass and Energy Releases for Containment Analyses, FSAR Chapter 6

The reduction in MMF flow can affect the mass and energy releases for containment analysis only through a change in the NC system temperature input assumption. RCS average temperature will remain unchanged with the change in MMF. Therefore, the RCS initial fluid and metal stored energy will remain unchanged. Further, a constant RCS average temperature implies that the driving temperature difference for primary to secondary heat transfer will remain unchanged. These two parameters, initial energy content and rate of energy transfer, are the means by which mass and energy releases influence containment response for the transients analyzed in Chapter 6 of the FSAR. Since the reduction in MMF is being made with a negligible change in RCS temperature, the mass and energy releases calculated in FSAR Chapter 6 will not be affected.

Accident Analyses, FSAR Chapter 15

All of the FSAR Chapter 15 accident analyses which are applicable to McGuire Nuclear Station and Catawba Nuclear Station have been explicitly analyzed with an initial RCS flow assumption which corresponds to a MMF of 382000 gpm, or have been evaluated to determine the impact of a reduction in MMF of 3000 gpm.

As shown in the updated FSAR, the following analyses have been analyzed with an initial RCS flow assumption which is less than or equal to a MMF flow of 382000 gpm. The results of the analyses demonstrate that all acceptance criteria are met, and therefore a MMF of 382000 gpm is acceptable:

15.1.5(1)	Steam System Piping Failure
15.2.3b	Turbine Trip - Peak Primary Pressure
15.2.6	Loss of Non-emergency AC Power
15.2.7	Loss of Normal Feedwater Flow
15.2.8	Feedwater System Pipe Break
15.3.1	Partial Loss of Reactor Coolant System Flow
15.3.2	Complete Loss of Reactor Coolant System Flow
15.3.3	Locked Rotor
15.4.1	Uncontrolled Bank Withdrawal from Subcritical
15.4.2(2)	Uncontrolled Bank Withdrawal at Power
15.4.3(2)	Rod Assembly Misoperation
15.4.8(1)	Rod Ejection
15.6.3(3)	Steam Generator Tube Rupture
15.6.5	Loss of Coolant Accident

Notes: 1) The updated FSAR Table 15-4 is incorrect for these events. The steam system piping failure, FSAR 15.1.5, and the rod ejection accident, FSAR

15.4.8, analyses have been submitted in Duke Power topical report DPC-NE-3001-PA. Table 15-4 for each station will be corrected in the next FSAR update.

2) The uncontrolled bank withdrawal at power, FSAR 15.4.2, and rod assembly misoperation, FSAR 15.4.3, events rely on cycle-specific reload analyses. Since the cycle specific analyses will be performed with a flow assumption of 382,000 gpm, FSAR Table 15-4 will be revised in the next FSAR update.

3) The steam generator tube rupture (SGTR), FSAR 15.6.3, event was inadvertently omitted from Table 15-4 of the updated FSAR. Table 15-4 of the Catawba Oct 91 FSAR update presented the correct input assumptions for the Catawba SGTR analysis. The McGuire SGTR analysis has no explicit flow assumption. Table 15-4 for each station will be corrected in the next FSAR update.

As stated in Duke Power Topical Report DPC-NE-3002-A, certain events are bounded by other more limiting events, and therefore are not analyzed and the results of these events are not affected by a change in MMF. The events which are bounded by other more limiting events are:

15.1.1	Reduction in Feedwater Temperature
15.1.4	Inadvertent Opening of a Steam Generator Relief Valve
15.2.2	Loss of External Load
15.2.4	Inadvertent Closure of Main Steam Isolation Valves
15.2.5	Loss of Condenser Vacuum and Events Causing Turbine Trip
15.3.4	Reactor Coolant Pump Shaft Break
15.5.1	Inadvertent Operation of ECCS
15.5.2	Increase in Reactor Coolant Inventory

The remaining Chapter 15 events which apply to McGuire and Catawba Nuclear Stations are events which are analyzed with the acceptance criterion of no DNB. These transients are non-limiting with respect to DNB, and DNB is not seriously challenged in any of these events. Therefore, a reduction in MMF of 3000 gpm is not significant to the results of the following analyses:

15.1.2	Increase in Feedwater Flow
15.1.3	Excessive Increase in Secondary Steam Flow
15.4.4	Startup of a Reactor Coolant Pump at an Incorrect Temperature

15.6.1 Inadvertent Opening of a Pressurizer Relief Valve

Conclusions

As shown above, all of the applicable FSAR analyses have been explicitly analyzed with an initial assumption which corresponds to a MMF of 382,000 gpm, or have been evaluated to determine the impact of a reduction in MMF of 3,000 gpm. Therefore, a decrease from 385000 gpm to 382000 gpm in the Catawba and McGuire Technical Specification minimum measured flow will not adversely affect the steady state or transient analyses documented in Chapters 3, 4, 6, and 15 of the Catawba and McGuire FSARs.

ATTACHMENT III

Analysis to Support the Conclusion of No Significant Hazard

The following analysis, performed pursuant to 10 CFR 50.91, shows that the proposed amendment will not create a significant hazards consideration as defined by the criteria of 10 CFR 50.92.

1. This amendment will not significantly increase the probability or consequence of any accident previously evaluated.

   No component modification, system realignment, or change in operating procedure will occur which could affect the probability of any accident or transient. The reduction in flow will not change the probability of actuation of any Engineered Safeguard Feature or other device. The consequences of previously-analyzed accidents have been found to be insignificantly different when the reduced flow rate is assumed. The system transient response is not affected by the initial RCS flow assumption, unless the initial assumption is so low as to impair the steady-state core cooling capability or the steam generator heat transfer capability. This is clearly not the case with a <1% reduction in RCS flow.

   The change to Technical Specification 2.1.1 to refer to DNB and CFT limits rather than Figure 2.1-1 will not cause the consequences of a previously analyzed accident to increase. No new mechanisms are introduced which could exacerbate a previously analyzed accident.

2. This amendment will not create the possibility of any new or different accidents not previously evaluated.

   No component modification, system realignment, or change in operating procedure will occur which could create the possibility of a new event not previously considered. The reduction in flow will not initiate any new events.

   The change to Specification 2.1.1 will not initiate any new events. The introduced T_{avg} and thermal power limits define a more restrictive operating range than could be inferred from the existing figure. There are no new mechanisms introduced which could create the possibility of a different accident not previously analyzed.

3. This amendment will not involve a significant reduction in a margin of safety.

   As described in Attachment II, the decrease in RCS flow has been analyzed and found to have an insignificant effect on the

applicable transient analyses found in the FSAR. In order to support the reduced flow rate, the OTΔT and OPΔT setpoint equation constants have been revised. There is no significant reduction in a margin of safety.

The change to Technical Specification 2.1.1 will not reduce a margin of safety. The limits on T_{avg} and thermal power will provide the reactor operator with meaningful and identifiable indications in the event that normal operating conditions are exceeded.