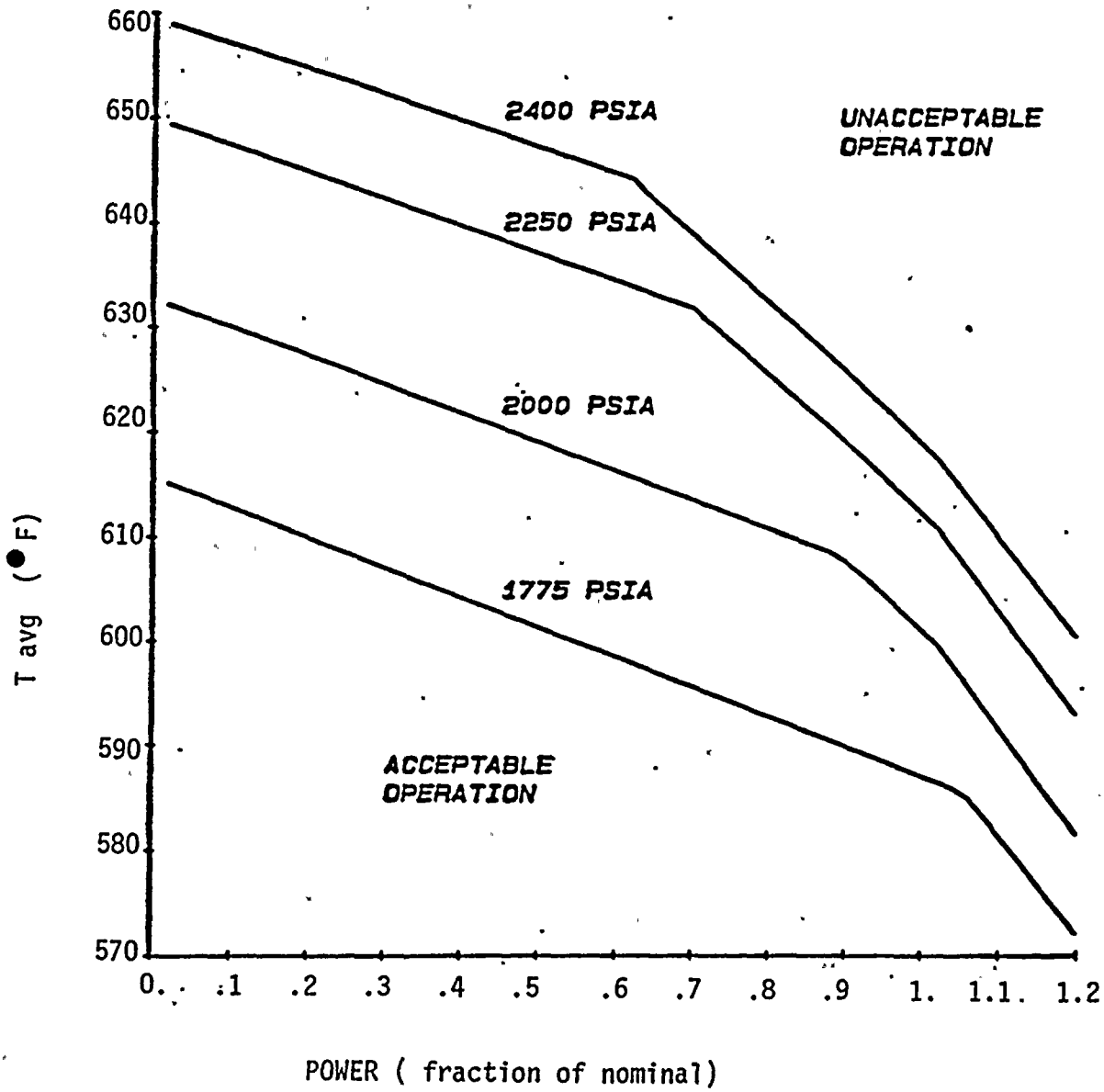
Attachment A

Revise the Technical Specification pages as follows:

Remove	Insert
p. 2.1-4	p. 2.1-4
p. 2.3-2	p. 2.3-2
p. 2.3-3	p. 2.3-3
p. 2.3-6 thru 2.3-9	p. 2.3-6 thru 2.3-9 (including 2.3-8a)

FIGURE 2.1-1
CORE DNB SAFETY LIMITS
2 LOOP OPERATION



Proposed

d. Overtemperature ΔT

$$\leq \Delta T_o \left[K_1 + K_2 (P-P^1) - K_3 (T-T^1) \left(\frac{1 + \tau 1S}{1 + \tau 2S} \right)\right] - f(\Delta I)$$

where

ΔT_o	=	indicated ΔT at rated power, °F
T	=	average temperature, °F
T^1	=	573.5°F
P	=	pressurizer pressure, psig
P^1	=	2235 psig
K_1	=	1.20
K_2	=	.000900
K_3	=	.0209
$\tau 1$	=	25 sec
$\tau 2$	=	5 sec

and f (ΔI) 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 where q_t and q_b are the percent power in the top and bottom halves of the core respectively, and $q_t + q_b$ is the total core power in percent of rated power such that:

(i) for $q_t - q_b$ less than +13 percent, $f(\Delta I) = 0$

(ii) for each percent that the magnitude of $q_t - q_b$ is more positive than +13 percent, the ΔT trip set point shall be automatically reduced by equivalent of 1.3 percent of rated power.

e. Overpower ΔT

$$\leq \Delta T_o \left[K_4 - K_5 (T-T^1) - K_6 \frac{\tau 3\ ST}{\tau 3S + 1}\right] - f(\Delta I)$$

where

ΔT_o	=	indicated ΔT at rated power, °F
T	=	average temperature, °F
T^1	=	indicated T avg at nominal conditions at rated power, °F
K_4	=	1.077
K_5	=	0.0 for $T<T^1$
	=	0.0011 for $T \geq T^1$
K_6	=	0.0262 for increasing T
	=	0.0 for decreasing T
$\tau 3$	=	10 sec
$f(\Delta I)$	=	as defined in 2.3.1.2.d

The overtemperature ΔT reactor trip provides core protection against DNB for all combinations of pressure, power, coolant temperature, and axial power distribution, provided only that: (1) the transient is slow with respect to the thermal capacity of the reactor coolant system to respond to power increases [1][2] and (2) pressure is within the range between the high and low pressure reactor trips. With normal axial power distribution, the reactor trip limit, with allowance for errors,[2] is always below the core safety limit as shown in Figure 2.1-1. If axial peaks are greater than design, as indicated by difference between top and bottom power range nuclear detectors, the reactor trip limit is automatically reduced.[4] The overpower ΔT reactor trip prevents power density anywhere in the core from exceeding a value at which fuel pellet centerline melting would occur as described in Section 7.2 of the UFSAR. This setpoint includes corrections for axial power distribution, change in density and heat capacity of water with temperature, and dynamic compensation for piping delays from the core to the loop temperature detectors. The specified set points meet this requirement and include allowance for instrument errors.[1] The low flow reactor trip protects the core against DNB in the event of a sudden loss of power to one or both reactor coolant pumps. The set points specified are consistent with the value used in the accident analysis.[1] The underfrequency reactor trip protects against a decrease in flow caused by low electrical frequency. The specified set point assures a reactor trip signal before the low flow trip point is reached.

The high pressurizer water level reactor trip protects the pressurizer safety valves against water relief. Approximately 700 ft.^{3} of water corresponds to 92% of span. A trip at this set point contains margin for both normal instrument error and transient overshoot of level beyond this trip setting. An additional 4% instrument error has been assumed to account for the effects of elevated temperatures on level measurement in accordance with IE Bulletin 79-21.(12) Therefore a trip setpoint of 88% prevents the water level from reaching the safety valves.(4)

The low-low steam generator water level reactor trip protects against loss of feedwater flow accidents. A set point of 5% is equivalent to at least 40,000 lbs. of water and assures that there will be sufficient water inventory in the steam generators at the time of trip to allow for starting delays for the auxiliary feedwater system.(5) An additional 11% has been added to the set point to account for error which may be introduced into the steam generator level system at a containment temperature of 286°F as determined by evaluation performed for temperature effects on level measurements required by IE Bulletin 79-21.

The specified reactor trips are blocked at low power where they are not required for protection and would otherwise interfere with normal plant operations. The prescribed set point above which these trips are unblocked assures their availability in the power range where needed.

Operation with one pump will not be permitted above 130 MWT (8.5%). An orderly power reduction to less than 130 MWT (8.5%) will be accomplished if a pump is lost while operating between 130 MWT (8.5%) and 50%. Automatic protection is provided so that a power-to-flow ratio is maintained equal to or less than one, which insures that the minimum DNB ratio increases at lower flow because the maximum enthalpy rise does not increase. For this reason the single pump loss of flow trip can be bypassed below 50% power.

The loss of voltage and degraded voltage trips ensure operability of safeguards equipment during a postulated design basis event concurrent with a degraded bus voltage condition.[9][10][11] The undervoltage set points have been selected so that safeguards motors will start and accelerate the driven loads (pumps) within the required time and will be able to perform for long periods of time at degraded conditions above the trip set points without significant loss of design life. All control circuitry or safety related control centers and load centers, except for motor control centers M and L, are d.c. Therefore, degraded grid voltages do not affect these control centers and load centers. Motor control centers M and L, which supply the Standby Auxiliary Feedwater System, are fully protected by the undervoltage set points. Further, the Standby System is normally not in service and is manually operated only in total loss of feedwater and auxiliary feedwater.

The 5% tolerance curve in Figure 2.3-1 and the requirements of specifications 2.3.3.1 and 2.3.3.2 include 5% allowance for measurement error. Thus, providing the measurement error is less than 5%, measured values may be directly compared to the curve. If measurement error exceeds 5%, appropriate allowance shall be made.

References:

(1) UFSAR 15.0

(2) UFSAR 15.4

(3) UFSAR 15.6

(4) UFSAR 7.2

(5) UFSAR 15.2

(6) Deleted

(7) Deleted

(8) Deleted

(9) Letter from L.D. White, Jr. to A. Schwencer, NRC, dated September 30, 1977.

(10) Letter from L.D. White, Jr. to A. Schwencer, NRC, dated September 30, 1977.

(11) Letter from L.D. White, Jr. to D. Ziemann, NRC, dated July 24, 1978.

(12) Letter from L.D. White, Jr. to B. Grier, USNRC dated September 14, 1979.

Attachment B

By submittal dated December 20, 1983 Rochester Gas and Electric provided to the NRC a Reload Transition Safety Report (RTSR) which identified the results of a reanalysis of FSAR Chapter 15 design basis events for the transition to a full core of Westinghouse OFA fuel. This analysis assumed a steam generator tube plugging level of 10 percent for non-LOCA events, and 12 percent for LOCA.

Because of anticipated increases in the percentage of plugged steam generator tubes in the Spring, 1988 refueling outage and beyond, RG&E requested that Westinghouse update the RTSR assuming steam generator tube plugging of 15 percent.

The scope and methodology of this reanalysis was discussed with the NRC Staff on August 11, 1987. The results are attached.

The following changes to the Technical Specifications are required:

1. Figure 2.1-1, Core DNB Safety Limits have been changed reflecting the 2.2 percent reduction in the RCS thermal design flow with the higher level of tube plugging.

2. The Overtemperature and Overpower Delta T setpoints have been changed to provide protection for the adjusted Core DNB Safety Limits.

3. Miscellaneous changes to the bases to incorporate an updated description and references.

These proposed changes have been reviewed against the standards of 10CFR50.92 to determine if they involve a significant hazards consideration. These changes do not involve a significant hazards consideration for the following reasons.

1. These changes will not result in a significant increase in the probability or consequences of an accident previously evaluated.

   The proposed changes involve the graphical depiction of core DNB safety limits and changes to OT and OP Delta T setpoints. These changes cannot affect the probability of occurrence of the accidents addressed in the FSAR and the RTSR referenced above.

   Westinghouse has evaluated the consequences of these accidents assuming 15 percent steam generator tube plugging. The results indicate that the differences between the RTSR previously approved and this analysis are minor with all established acceptance criteria satisfied. With these established criteria satisfied, the consequences of the accident will not be increased.

2. These changes will not create the possibility of a new or different kind of accident from any previously analyzed.

   As indicated above the proposed changes to the Technical Specification do not involve a physical modification to the Plant that could result in the creation of an accident not previously analyzed.

3. These changes do not involve a significant reduction in the margin of safety.

   The attached Westinghouse report documents the reevaluation - reanalysis of the events described in Chapter 15 of the UFSAR.

   This report compares the results of the analysis with the established acceptance criteria for each design basis event. The acceptance criteria incorporates conservatisms such that by satisfying the criteria the established margins of safety are maintained. Therefore, there are no reductions in the margins of safety.