ATTACHMENT I

PROPOSED TECHNICAL SPECIFICATIONS CHANGES

RELATED TO

SAFETY LIMIT, REACTOR CORE

AND

CONTROL ROD AND POWER DISTRIBUTION LIMITS,

SHUTDOWN REACTIVITY

POWER AUTHORITY OF THE STATE OF NEW YORK
INDIAN POINT 3 NUCLEAR POWER PLANT
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thermal power, reactor coolant temperature and pressure have been related to DNB through the W-3 DNB "L" grid geometry correlation.[3] The W-3 DNB correlation 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 minimum value of the DNBR during steady state operation, normal operational transients, and anticipated transients is limited to 1.30. This corresponds to a 95% probability at a 95% confidence level that DNB will not occur and is chosen as an appropriate margin to DNB for all operating conditions.[1]

The curves of Figures 2.1-1 and 2.1-2 represent the loci of points of thermal power, coolant system pressure and average temperature for which the DNBR is no less than 1.30. The area where clad integrity is assured is below these lines.

The calculation of these limits includes an $F^N_{\Delta H}$ of 1.55, DNB penalties for increased pellet eccentricity, local power spikes, 8% uncertainty in $F^N_{\Delta H}$, up to 12% steam generator tube plugging, and a reference cosine with a peak of 1.55 for axial power shape.[3]

Figures 2.1-1 and 2.1-2 include an allowance for an increase in the enthalpy rise hot channel factor at reduced power based on the expression:

$F^N_{\Delta H} = 1.55\ [1 + 0.2\ (1-P)]$ where P is the fraction of rated power.[3]

The control rod insertion limits are covered by Specification 3.10. Higher hot channel factors could occur at lower power levels because additional control rods are in the core. However, the control rod insertion limits for four loop and three loop operation as dictated by Figures 3.10-4 and 3.10-5, respectively, insure that the DNBR is always greater at partial power than at full power.[3]

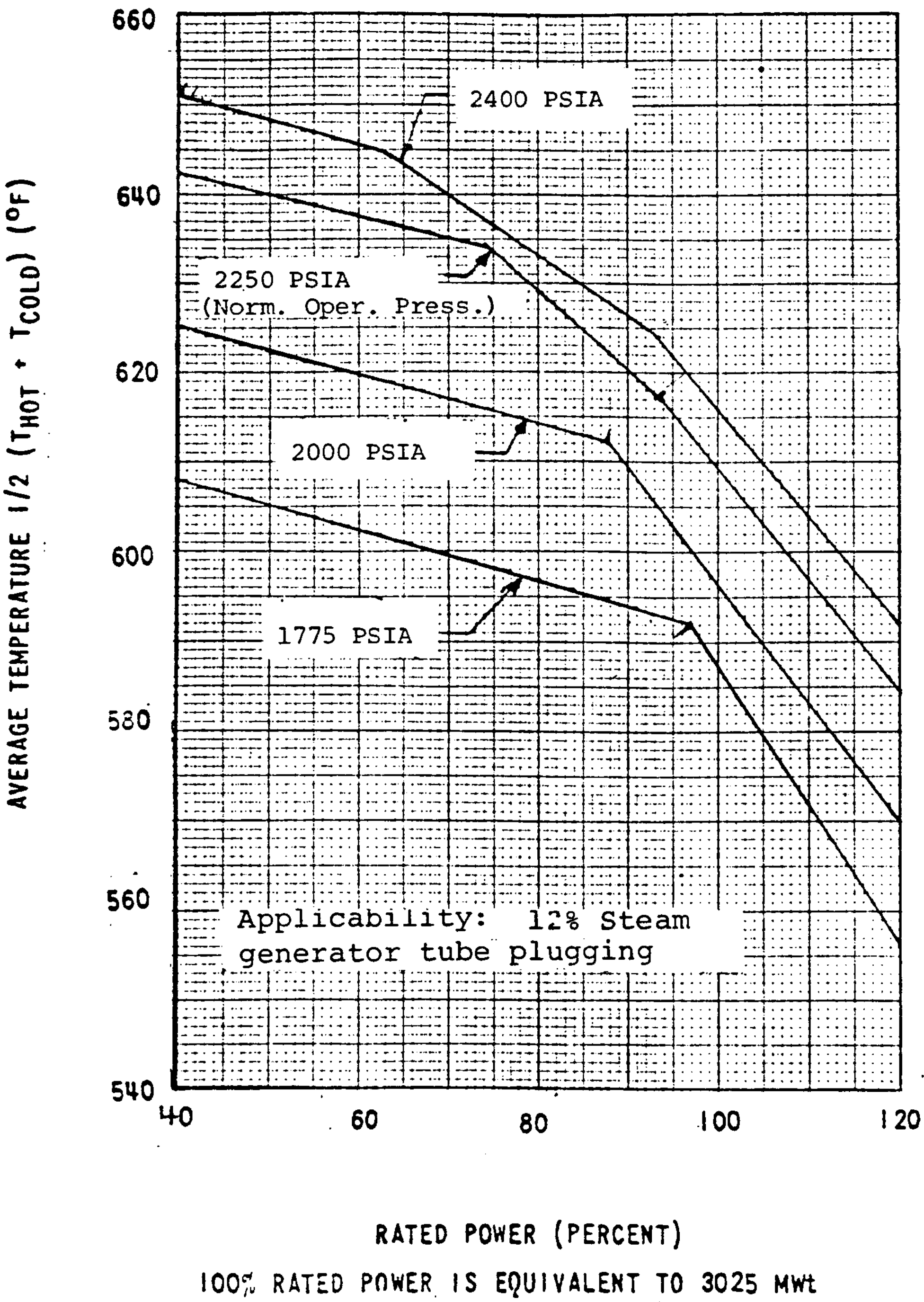

Figure 2.1-1. Core Limits - Four Loop Operation

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where

ΔT_o = Indicated ΔT at rated power, 57.8°F

T_{avg} = Average temperature, °F

T' = Indicated T_{avg} at nominal conditions at rated power, 571.5°F

P = Pressurizer pressure, psig

P' = Indicated nominal pressurizer pressure at rated power = 2235 psig

$\left.\begin{array}{l} K_1 \leq 1.200 \\ K_2 \geq 0.0129 \\ K_3 \leq 0.00073 \end{array}\right\}$ Four Loop Operation

$\left.\begin{array}{l} K_1 \leq 1.110 \\ K_2 \geq 0.0129 \\ K_3 \leq 0.00073 \end{array}\right\}$ Three Loop Operation

K_1 is a constant which defines the over temperature ΔT trip margin during steady state operation if the temperature, pressure and $f(\Delta I)$ terms are zero.

K_2 is a constant which defines the dependence of the overtemperature ΔT set point to T_{avg}.

K_3 is a constant which defines the dependence of the overtemperature ΔT set point to pressurizer pressure.

ΔI = $q_t - q_b$, 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 total core power in percent of rated power.

$f(\Delta I)$ = 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 as defined above such that:

(a) for $q_t - q_b$ within -20, +15 percent, $f(\Delta I) = 0$.

(b) for each percent that the magnitude of $q_t - q_b$ exceeds +15 percent, the ΔT trip set point shall be automatically reduced by an equivalent of 6.0 percent of rated power.

(c) for each percent that the magnitude of $q_t - q_b$ exceeds -20 percent, the ΔT trip setpoint shall be automatically reduced by an equivalent of 1.5 percent of rated power.

3.10 CONTROL ROD AND POWER DISTRIBUTION LIMITS

Applicability:

Applies to the limits on core fission power distributions and to the limits on control rod operations.

Objectives:

To ensure:

1. Core subcriticality after reactor trip,
2. Acceptable core power distribution during power operation in order to maintain fuel integrity in normal operation and transients associated with faults of moderate frequency, supplemented by automatic protection and by administrative procedures, and to maintain the design basis initial conditions for limiting faults, and
3. Limit potential reactivity insertions caused by hypothetical control rod ejection.

Specifications:

3.10.1 Shutdown Reactivity

The shutdown margin shall be at least as great as shown in Figure 3.10-1.

3.10.2 Power Distribution Limits

3.10.2.1 At all times, except during low power physics tests, the hot channel factors defined in the basis must meet the following limits:

$$F_Q(Z) \leq (2.04/P) \times K(Z) \text{ for } P > 0.5$$
$$F_Q(Z) \leq (4.08/P) \times K(Z) \text{ for } P \leq 0.5$$
$$F^N_{\Delta H} \leq 1.55\ [1 + 0.2\ (1-P)]$$

where P is the fraction of full power at which the core is operating. K(Z) is the fraction given in Figure 3.10-2 and Z is the core height location of F_Q.

F_Q^E, Engineering Heat Flux Hot Channel Factor, is defined as the allowance on heat flux required for manufacturing tolerances. The engineering factor allows for local variations in enrichment, pellet density and diameter, surface area of the fuel rod and eccentricity of the gap between pellet and clad. Combined statistically the net effect is a factor of 1.03 to be applied to fuel rod surface heat flux.

$F_{\Delta H}^N$, 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.

It should be noted that $F_{\Delta H}^N$ is based on an integral and is used as such in the DNB calculations. Local heat fluxes are obtained by using hot channel and adjacent channel explicit power shapes which take into account variations in horizontal (x-y) power shapes throughout the core. Thus the horizontal power shape at the point of maximum heat flux is not necessarily directly related to $F_{\Delta H}^N$.

An upper bound envelope of 2.04 times the normalized peaking factor axial dependence of Figure 3.10-2 has been determined consistent with Appendix K criteria and is satisfied by all operating maneuvers consistent with the technical specifications on power distribution control as given in Section 3.10. The results of the loss of coolant accident analyses based on this upper bound normalized envelope of Figure 3.10-2 demonstrate a peak clad temperature of 1996°F, which is well within the peak clad temperature limit of 2200°F. [2]

When an F_Q 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 moveable incore detector flux mapping system and three percent is the appropriate allowance for manufacturing tolerance.

In the specified limit of $F_{\Delta H}^N$ there is a 8 percent allowance for uncertainties which means that normal operation of the core is expected to result in $F_{\Delta H}^N \leq 1.55/1.08$. The logic behind the larger uncertainty in this case is that (a) normal perturbations in the radial power shape

4. Axial power distribution control procedures, which are written in terms of flux difference control and control bank insertion limits, are observed. Flux difference refers to the difference in signals between the top and bottom halves of two-section excore neutron detectors. The flux difference is a measure of the axial offset which is defined as the difference in normalized power between the top and bottom halves of the core.

The permitted relaxation in $F_{\Delta H}^{N}$ allows radial power shape changes with rod insertion to the insertion limits. It has been determined that provided the above conditions 1 through 4 are observed, these hot channel factors limits are met. In Specification 3.10.2, F_Q is arbitrarily limited for $P \leq 0.5$ (except for low power physics tests).

The procedures for axial power distribution control referred to above are designed to minimize the effects of xenon redistribution on the axial power distribution during load-follow maneuvers. Basically, control of flux difference is required to limit the difference between the current value of Flux Difference (ΔI) and a reference value which corresponds to the full power equilibrium value of Axial Offset (Axial Offset = ΔI/fractional power). The reference value of flux difference varies with power level and burnup but expressed as axial offset it varies only with burnup.

The technical specifications on power distribution control assure that F_Q upper bound envelope of 2.04 times Figure 3.10-2 is not exceeded and xenon distributions are not developed which at a later time, would cause greater local power peaking even though the flux difference is then within the limits specified by the procedure.

The target (or reference) value of flux difference is determined as follows. At any time that equilibrium xenon conditions have been established, the indicated flux difference is noted with

the control rod bank more

than 190 steps withdrawn (i.e. normal full power operating position appropriate for the time in life, usually withdrawn farther as burnup

ATTACHMENT II

SAFETY EVALUATION

RELATED TO

SAFETY LIMIT, REACTOR CORE

AND

CONTROL ROD AND POWER DISTRIBUTION LIMITS,

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Section I - Description of Modification

The proposed changes to the Technical Specifications are shown in Attachment I. Sections 2.1, 2.3 and 3.10 of the Technical Specifications have been revised. Also, the Section 3.10 Basis has been revised. These proposed changes result from the ECCS reanalysis which modifies the reactor core limits and the heat flux peaking factor (F_Q) value. The Westinghouse ECCS reanalysis report for steam generator tube plugging level of twelve (12) percent is enclosed as Attachment III to this submittal.

Section II - Purpose of Modification

The purpose of the modification is to revise the IP-3 Technical Specifications so as to comply with the current ECCS reanalysis.

Section III - Impact of the Change

The proposed changes to the Technical Specifications do not change any system or subsystem. The impact is to permit plant operations with a higher percentage of steam generator tubes plugged. This requires a lower heat flux peaking factor value to meet peak clad temperature requirements.

Section IV - Implementation of the Modification

The modification as proposed will not impact the ALARA or Fire Protection Program at IP-3.

Section V - Conclusion

The incorporation of these modifications: a) will not change the probability nor the consequences of an accident or malfunction of equipment important to safety as previously evaluated in the Safety Analysis Report; b) will not increase the possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report; c) will not reduce the margin of safety as defined in the basis for any Technical Specificaton; and d) does not constitute an unreviewed safety question.

Section VI - References

(a) IP-3 FSAR

(b) IP-3 SER

ATTACHMENT III

EMERGENCY CORE COOLING SYSTEM REANALYSIS
FOR STEAM GENERATOR TUBE PLUGGING
LEVEL OF TWELVE PERCENT

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