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Docket No. 50-286

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Mr. J. P. Bayne
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White Plains, New York 10601

Dear Mr. Bayne:

Subject: Steam Generator Tube Plugging at the Indian Point Nuclear Generating Plant, Unit No. 3 (IP-3)

By letter dated January 13, 1984, we issued License Amendment No. 48 to Facility Operating License No. DPR-64 for IP-3 regarding the above subject. This amendment included revised Technical Specification pages some of which were illegible.

Enclosed is a complete set of clear Technical Specification pages. These pages replace the pages previously forwarded with Amendment No. 48.

Sincerely,

/s/ Philip J. Polk

Philip J. Polk, Project Manager
Operating Reactors Branch No. 1
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Enclosures:
Attachment Sheet and Revised
Technical Specification pages

cc w/enclosures:
See attached list

| | | |
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ATTACHMENT TO LICENSE AMENDMENT NO. 48

FACILITY OPERATING LICENSE NO. DPR-64

DOCKET NO. 50-286

Revise Appendix A as follows:

| Remove Pages | Insert Pages |
|---|---|
| 2.1-1 | 2.1-1 |
| 2.1-2 | 2.1-2 |
| Figure 2.1-1 | Figure 2.1-1 |
| Figure 2.1-2 | Figure 2.1-2 |
| 2.3-2 | 2.3-2 |
| 2.3-3 | 2.3-3 |
| 3.1-1 | 3.1-1 |
| 3.1-3 | 3.1-3 |
| 3.10-1 | 3.10-1 |
| 3.10-9 | 3.10-9 |
| 3.10-11 | 3.10-11 |
| Figure 3.10-5 | Figure 3.10-5 |

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# 2 SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

## 2.1 Safety Limit, Reactor Core

### Applicability

Applies to the limiting combinations of thermal power, Reactor Coolant System pressure and coolant temperature during four-loop operation.

### Objective

To maintain the integrity of the fuel cladding.

### Specification

The combination of thermal power level, coolant pressure, and coolant temperature shall not exceed the limits shown in Figure 2.1-1 for four-loop operation. The safety limit is exceeded if the point defined by the combination of Reactor Coolant System average temperature and power level is at any time above the appropriate pressure line.

### Basis

To maintain the integrity of the fuel cladding and prevent fission product release, it is necessary to prevent overheating of the cladding under all operating conditions. This is accomplished by operating the hot region of the core within the nucleate boiling regime of heat transfer, wherein the heat transfer coefficient is very large and the clad surface temperature is only a few degrees Fahrenheit above the coolant saturation temperature. The upper boundary of the nucleate boiling regime is termed departure from nucleate boiling (DNB) and at this point there is a sharp reduction of the heat transfer coefficient, which would result in high clad temperatures and the possibility of clad failure. DNB is not, however, an observable parameter during reactor operation. Therefore, the observable parameters:

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.

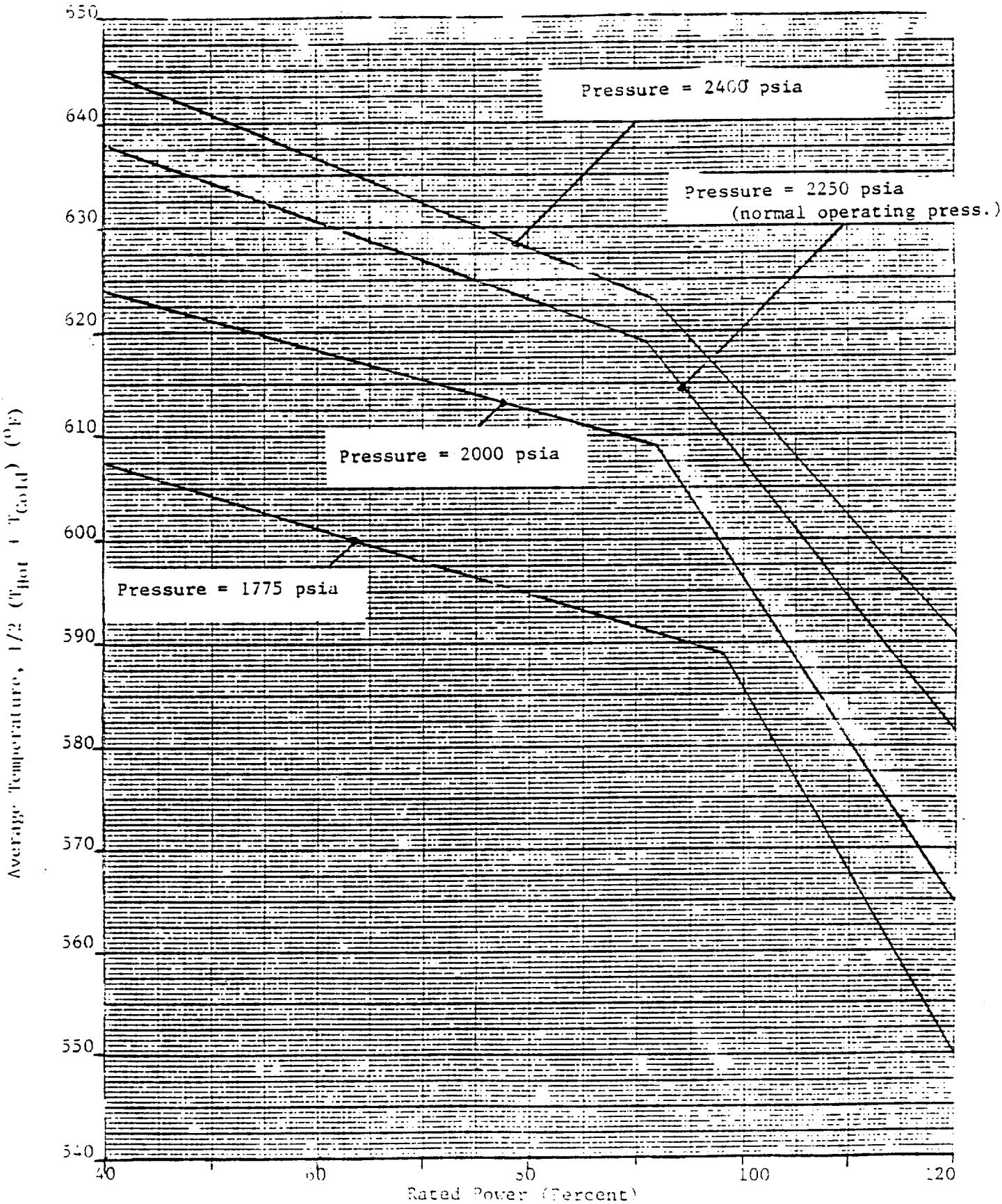
The calculation of these limits includes an $F^N_{\Delta H}$ of 1.55, DNB penalties for increased pellet eccentricity, local power spikes, 8% uncertainty in $F^N_{\Delta H}$, up to 24% maximum tube plugging level for each steam generator and a reference cosine with a peak of 1.55 for axial power shape.[3]

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]

Reactor Coolant Flow > 323600 G
24 Percent Tube Plugging




100 Percent Rated Power is Equivalent to 3025 $MW_{th}$

Amendment No. 48

Figure 2.1-1

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Figure 2.1-2. Core Limits - Three Loop Operation

$\Delta T_o$ = Indicated $\Delta T$ at rated power, $\leq 63.5°F$

$T_{avg}$ = Average Temperature, °F

$T'$ = Indicated $T_{avg}$ at nominal conditions at rated power, 574.7° F

$P$ = Pressurizer pressure, psig

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

$K_1$ $\leq$ 1.135

$K_2$ = $0.0114 \pm .00057$

$K_3$ = $0.00066 \pm .0000033$

$K_1$ is a constant which defines the over temperature $\Delta T$ 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 $\Delta T$ set point to $T_{avg}$

$K_3$ is a constant which defines the dependence of the overtemperature $\Delta T$ set point to pressurizer pressure.

$\Delta I$ = $q_t - q_b$, 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 defined above such that:

(a) for $q_t - q_b$ within -20, + 10 percent, $f(\Delta I) = 0$.

(b) for each percent that the magnitude of $q_t - q_b$ exceeds + 10 percent, the $\Delta T$ 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 $\Delta T$ trip setpoint shall be automatically reduced by an equivalent of 1.5 percent of rated power.

(5) Overpower $\Delta T$

$$\Delta T \leq \Delta T_o \left[K_4 - K_5 \frac{dT_{avg}}{dt} - K_6 (T_{avg} - T') - f(\Delta I)\right]$$

where

$\Delta T_o$ = indicated $\Delta T$ at rated power, (100% full power measured $\Delta T$, no greater than 63.5° F.)

$T_{avg}$ = average temperature, °F

$T'$ = indicated $T_{avg}$ at nominal conditions at rated power, $\leq$ 574.7°F

$K_4$ $\leq$ 1.089

$K_5$ = 0 for decreasing average temperature

$\geq$ 0.175 sec/°F for increasing average temperature

$K_6$ = 0 for $T \leq T'$

$\geq$ 0.00116 for $T > T'$

$K_4$ is a constant which defines the overpower $\Delta T$ trip margin during steady state operation if the temperature and the $f(\Delta I)$ terms are zero.

$K_5$ is a constant determined by dynamic considerations to compensate for piping delays from the core to the loop temperature detectors; it represents the combination of the equipment static gain setting and the time constant setting.

$K_6$ is a constant which defines the dependence of the overpower $\Delta T$ setpoint to $T_{avg}$

$f(\Delta I)$ = as defined above.

$\frac{dT_{avg}}{dt}$ = rate of change of $T_{avg}$

(6) Low reactor coolant loop flow:

(a) $\geq$ 90% of normal indicated loop flow

(b) Low reactor coolant pump freqency - $\geq$55.0 cps

(7) Undervoltage - $\geq$70% of normal voltage

For the cases where no exception time is specified for inoperable components, this time is assumed to be zero.

## 3.1 REACTOR COOLANT SYSTEM

### Applicability

Applies to the operating status of the Reactor Coolant System; operational components; heatup, cooldown, criticality, activity, chemistry and leakage.

### Objective

To specify those limiting conditions for operation of the Reactor Coolant System which must be met to ensure safe reactor operation.

### Specification

A. OPERATIONAL COMPONENTS

1. Coolant Pumps

   a. At least one reactor coolant pump or one residual heat removal pump in the Residual Heat Removal System when connected to the Reactor Coolant System shall be in operation when a reduction is made in the boron concentration of the reactor coolant.

   b. When the reactor is critical and above 2% rated power, except for natural circulation tests, at least two reactor coolant pumps shall be in operation.

   c. The reactor shall not be operated at power levels above 10% rated power with less than four (4) reactor coolant loops in operation.

Basis

When the boron concentration of the Reactor Coolant System is to be reduced the process must be uniform to prevent sudden reactivity changes in the reactor. Mixing of the reactor coolant will be sufficient to maintain a uniform boron concentration if at least one reactor coolant pump or one residual heat removal pump is running while the change is taking place. The residual heat removal pump will circulate the primary system volume in approximately one half hour. The pressurizer is of no concern because of the low pressurizer volume and because the pressurizer boron concentration will be higher than that of the rest of the reactor coolant.

Heat transfer analyses show that reactor heat equivalent to 10% of rated power can be removed with natural circulation only (1); hence, the specified upper limit of 2% rated power without operating pumps provides a substantial safety factor.

The reactor shall not be operated at power levels above 10% rated power with less than four (4) reactor coolant loops in operation until safety analyses for less than four loop operation have been submitted by the licensee and approval for less than four loop operation at power levels above 10% rated power has been granted by the Commission. (See license condition 2.C.(3))

Each of the pressurizer code safety valves is designed to relieve 420,000 lbs. per hr. of saturated steam at the valve set point.

If no residual heat were removed by the Residual Heat Removal System the amount of steam which could be generated at safety valve relief pressure would be less than half the capacity of a single valve. One valve therefore provides adequate protection for overpressurization.

The combined capacity of the three pressurizer safety valves is greater than the maximum surge rate resulting from complete loss of load (2) without a direct reactor trip or any other control.

The requirement that 150 kw of pressurizer heaters and their associated controls be capable of being supplied electrical power from an emergency bus provides assurance that these heaters can be energized during a loss of offsite power condition to maintain natural circulation at hot shutdown.

The power operated relief valves (PORVS) operate to relieve RCS pressure below the setting of the pressurizer code safety valves. These relief valves have remotely operated block valves to provide a positive shutoff capability should a relief valve become inoperable. The electrical power for both the relief valves and the block valves is capable of being supplied from an emergency power source to ensure the ability to seal off possible RCS leakage paths.

References

1) FSAR Section 14.1.6

2) FSAR Section 14.1.8

## 3.10 CONTROL ROD A POWER DISTRIBUTION LIMITS

### Applicability:

Applies to the limits on core fission power distributions and to 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 main tain 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:

$$F_Q(Z) \leq (2.14/p) \times K(Z) \text{ for } P > 0.5$$

$$F_Q(Z) \leq (4.28) \quad K(Z) \text{ for } \leq 0.5$$

$$F^N_{\Delta H} \leq 1.55\ [1 + 0.2\ (1-P)]$$

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 2.14 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 1995°F, which is below 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 given 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 botton halves of the core.

The permitted relaxation in $F^N_{\Delta H}$ 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 ($\Delta I$) and a reference value which corresponds to the full power equilibrium value of Axial Offset (Axial Offset = $\Delta I$/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 evelope of 2.14 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 differnece 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

D E L E T E D

Figure 3.10-5 Insertion Limits 100 Step
Overlap 3 Loop Operation

Amendment No. ~~14~~, ~~40~~, 48