Docket No. 50-286

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Mr. J. P. Bayne
Executive Vice President
Nuclear Generation
Power Authority of the State
of New York
123 Main Street
White Plains, New York 10601

January 13, 1984

Dear Mr. Bayne:

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

The Commission has issued the enclosed Amendment No. 48 to Facility Operating License No. DPR-64 for the Indian Point Nuclear Generating Plant Unit No. 3. The amendment allows operation with a maximum of 24% steam generator tubes plugged in any steam generator.

The enclosed amendment consists of changes to the Technical Specifications in response to your submittal dated January 13, 1983, as augmented by your submittal dated December 14, 1983. The December letter was submitted in response to an NRC request for additional information. This information did not significantly change the original amendment request of January 1983.

By license Amendment Number 40 to DPR-64, dated November 13, 1981, 12% steam generator plugging was approved. The enclosed amendment supercedes the 12% approval and, as mutually agreed to by members of your staff, increases allowable tube plugging to a maximum of 24% for each steam generator. Approval is for fuel cycle 3. IP-3 has been granted 10 CFR 50.59 reload approval. As such, a 10 CFR 50.59 evaluation is required for the upcoming fuel cycle 4. NRC Region I has been requested to review your evaluation prior to restart.

A copy of the related Safety Evaluation is enclosed. The Notice of Issuance will be included in the Commission's next regular monthly Federal Register notice.

Sincerely,

Philip J. Polk

Philip J. Polk, Project Manager
Operating Reactors Branch No. 1
Division of Licensing

Enclosures:
1. Amendment No. to DPR-64
2. Safety Evaluation

cc w/enclosures:
See attached list

ORB #1	ORB #1	ORB #1	OELD	AD:DL
CParrish/jm	P Polk	S Varga		G Lainas
1/12/84	1/12/84	1/12/84	1/12/84	1/12/84

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

January 13, 1984

POWER AUTHORITY OF THE STATE OF NEW YORK

DOCKET NO. 50-286

INDIAN POINT NUCLEAR GENERATING UNIT NO. 3

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 48
License No. DPR-64

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Power Authority of the State of New York (the licensee) dated January 13, 1983, as supplemented December 14, 1983, 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.

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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. DPR-64 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 48 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

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

FOR THE NUCLEAR REGULATORY COMMISSION

Steven A. Varga, Chief
Operating Reactors Branch No. 1
Division of Licensing

Attachment:
Changes to the Technical
Specifications

Date of Issuance: January 13, 1984

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

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.

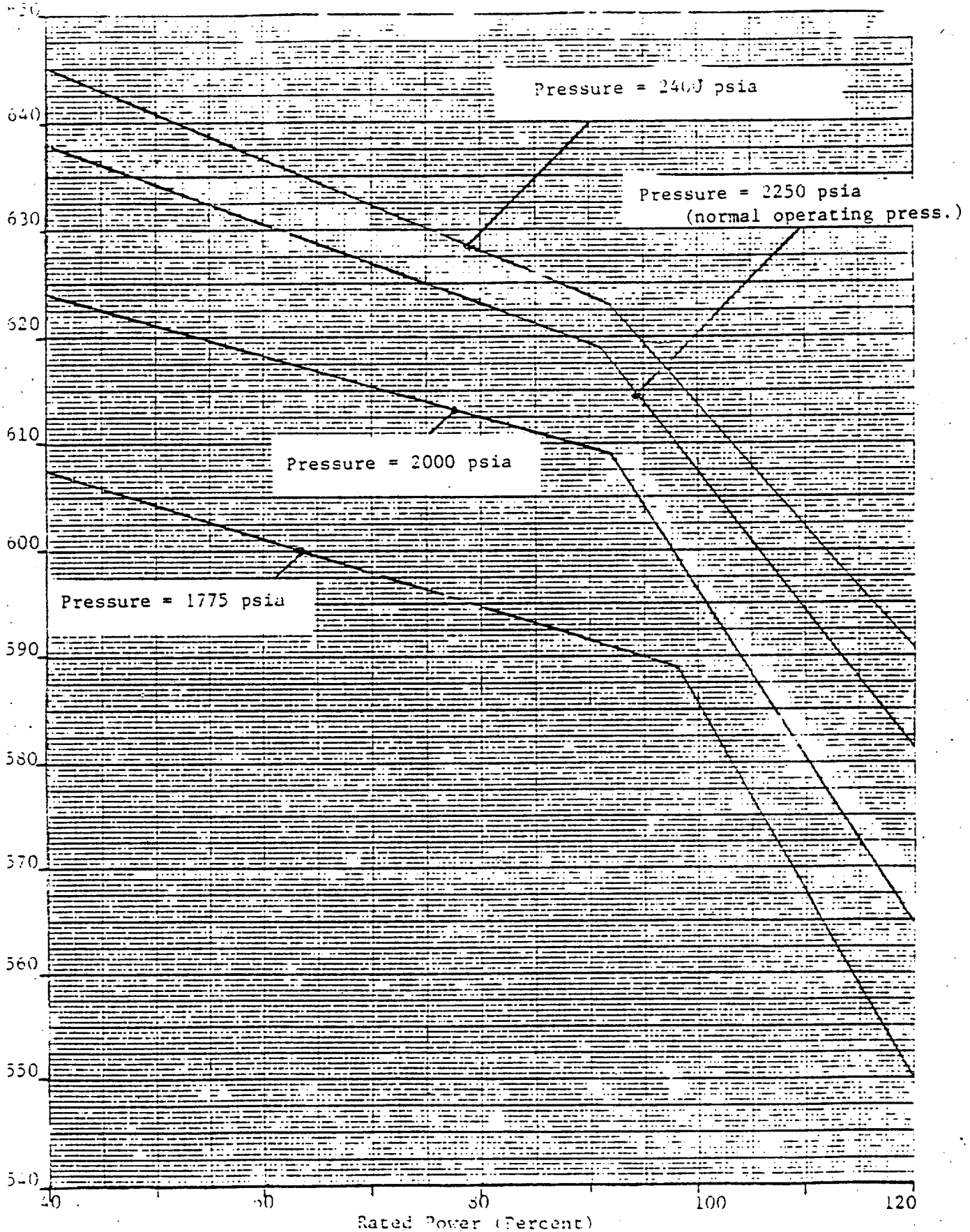
Specification

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:

Reactor Coolant Flow > 123600 GPM
24 Percent Tube Plugging



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

Amendment No. 48

Figure 2.1-1

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]

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Figure 2.1-2. Core Limits - Three Loop Operation

ΔT_o = Indicated ΔT at rated power, $\leq 63.5^oF$

T_{avg} = Average Temperature, ^oF

T' = Indicated T_{avg} at nominal conditions at rated power, 574.7^oF

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 ΔT trip margin during steady state operation if the temperature, pressure and f (ΔI) 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, + 10 percent, $f(\Delta I) = 0$.

(b) for each percent that the magnitude of $q_t - q_b$ exceeds + 10 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.

(5) Overpower ΔT

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

where

ΔT_o = indicated ΔT at rated power, (100% full power measured ΔT, 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 ΔT 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 ΔT 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 frequency — $\geq$ 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

Applicability:

Applies to the limits on core fiss. 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:

$$F_Q(Z) \leq (2.14/P) \times K(Z) \text{ for } P > 0.5$$
$$F_Q(Z) \leq (4.28) \times K(Z) \text{ for } P \leq 0.5$$
$$F^N_{\Delta H} \leq 1.55\ [1 + 0.2\ (1-P)]$$

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.

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.

$F^N_{\Delta H}$, 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 $F^N_{\Delta H}$ 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 $F^N_{\Delta H}$.

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

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 2.14 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 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

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Figure 3.10-5 Insertion Limits 100 Step

Overloop 3 Loop Operation

Amendment No. 48

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 48 TO FACILITY OPERATING LICENSE NO. DPR-64

POWER AUTHORITY OF THE STATE OF NEW YORK

INDIAN POINT NUCLEAR GENERATING UNIT NO. 3

DOCKET NO. 50-286

I. Background

The licensee (NYPA) has applied for approval to operate Indian Point Unit 3 with up to 24% steam generator tube plugging (Reference 1). The current limit is 12%. This amendment also entails an increase in core average temperature from 571.5°F to 574.7°F and an increase in ΔT from 57.8°F to 63.5°F. Furthermore, the OTΔT parameters would all be changed and there would be a small change in the OPΔT parameters, K_4 and K_6.

The amendment would also allow an increase in the peaking factor F_q from 2.04 to 2.14.

In support of these changes, the licensee submitted a complete reanalysis of the design basis LBLOCA (Reference 1). Subsequently, in response to a request from NRC, the licensee also submitted a reevaluation of non-LOCA transients and accidents (Reference 2).

II. Evaluation

The proposed changes in OTΔT and OPΔT setpoints were calculated using a standard computer code (OPTOAX), and are in general more limiting than the previous values during steady state operation. These values in conjunction with the transient analysis described below provide MDNBR protection. We conclude that the new values are acceptable.

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All of the other proposed changes relate to plant operating parameters, and must be justified based on a full set of transient and accident calculations and evaluations. The analyses provided by the licensee are described and evaluated below.

A. - Loss Of Coolant Accident (LOCA)

The licensee analyzed the Large Break LOCA for the full range of discharge coefficients, using the currently approved 1981 version of the Westinghouse evaluation model. For the worst case value of C_D = 0.4, assuming an F_q equal to 2.14, the calculated peak clad temperature (PCT) was below 2000°F, and the other acceptance criteria of 10 CFR 50 Appendix K were also met. Indian Point is limited to a calculated PCT of 2000°F instead of the 2200°F criterion of Appendix K.

The LBLOCA calculations were performed with an approved code and yielded results which meet the criteria of Appendix K. Therefore the analysis is acceptable. An unusual aspect of the calculations is the fact that no reduction in F_q was required to compensate for the increased tube plugging. Instead, the results allow both the percentage of tube plugging and the peaking factor to increase substantially over the previous LBLOCA analysis. This is due to the fact that the previous results embody substantial penalties due to an interim method of accounting for clad ballooning and rupture. The new value of F_q equal to 2.14 is still low compared with other plants of this type.

The Small Break LOCA was not reanalyzed for the proposed new conditions. The current SBLOCA calculation for Indian Point 3 was performed with a much higher peaking factor (F_q=2.32) and shows a margin of 200°F to the allowed peak clad temperature of 2000°F. Given the conservative value of F_q used in that calculation, the large margin to the PCT limit and the relative insensitivity of SBLOCA to changes in steam generator tube plugging, we concur with the licensee's assessment that detailed reanalysis of SBLOCA is not needed.

B. - Non-LOCA Transients And Accidents

Complete reanalysis of non-LOCA transients and accidents has not always been required to justify increases in the percentage of steam generator tube plugging. In general, the LBLOCA analysis for increased SGTP leads to a compensating decrease in F_q, and this result is judged to bound the non-LOCA events. In such cases, the licensee has been required only to qualitatively reevaluate the non-LOCA events. As discussed above, the current submittal is different insofar as the LBLOCA analysis allows both a substantially higher (less conservative) F_q and a substantially higher (less conservative) percentage of tube plugging than the previous analysis. This combination of increased plugging and peaking factor, coupled with the higher core average temperature and increased ΔT, prompted the staff to require a full reexamination of non-LOCA transients and accidents. The results are presented and evaluated below.

1. Accidents Not Reanalyzed: A number of accident scenarios normally discussed in an FSAR were not recalculated as a result of the proposed changes. This included accidents which are virtually independent of coolant flow, heat transfer and other thermal-hydraulic parameters; and accidents in which the decreased heat transfer due to tube plugging would be a benefit (MSLB). Malfunctions of the CVCS (boron dilution) were shown to be only marginally affected by the reduction in RCS volume, and this change is more than compensated by the fact that the shutdown margin (1.72% ΔK/K) is much higher than assumed in the FSAR (1.0%). Uncontrolled RCCA withdrawal from subcritical is impacted by the reduced core flow, resulting in faster fuel temperature rise. The course of this event is dominated by the Doppler Coefficient which renders the transient insensitive to the impact of steam generator tube plugging.

2. Loss Of Load: The loss-of-external-electrical-load transient is one of several events in which heat removal from the primary system is disrupted, resulting in an increase in RCS pressure and/or a decrease in DNBR. Such accidents would be aggravated by the additional tube plugging, due to reduced heat transfer area and reduced flow. Furthermore, the initial value of DNBR would be decreased because of the higher core exit temperature and higher power peaking factor. The reanalysis of the loss

of load transient included the same four cases analyzed in the FSAR; beginning of life and end of life reactivity feedback, with and without pressurizer pressure control. The peak pressures achieved were generally higher as expected, but in all cases, pressure remained well below 110% of design pressure. Because of the rapid pressure rise, DNBR does not fall below 1.30 for this transient.

3. Primary Coolant Flow Transients

Two accidents in this category were reanalyzed; simultaneous coastdown of all four coolant pumps and the single locked rotor event.

Simultaneous coastdown of all reactor coolant pumps causes a rapid increase in coolant temperatures and a consequent reduction in DNBR. The FSAR analysis of this event shows a decrease in DNBR from the initial value of 1.88 to a minimum of 1.63 at 2.5 seconds into the event. With 24% tube plugging, DNBR drops from the initial valve of 1.60 to a low of about 1.42 at 2.5 seconds. While the calculated minimum DNBR is lower in the new analysis, it is still well above the allowed minimum of 1.30 for this event.

The locked rotor accident results in a rapid pressure increase, a similarly rapid rise in clad temperatures, and a drop in DNBR. This accident is less severe for

4-loop plants than for others, because the lost flow is a smaller fraction of the total. The FSAR calculates a peak pressure of 2417 psia, a peak clad temperature of 1665°F and a minimum DNBR under 1.30 for less than 1% of the rods. The new analysis calculates a somewhat higher pressure, 2515 psia, but still well within the acceptance criterion. The peak clad temperature of 1849°F is theoretically high enough to allow clad-coolant interactions but is of sufficiently short duration (3-6 seconds) that fuel damage propogation is not expected. A minimum DNBR of less than 1.30 is estimated to occur for 0.5% of the rods. This level of rod failure would not exceed the dose rate criteria.

4. Reactivity Transients

The control rod related transient events are not significantly altered directly by the changes produced by tube plugging. The reactivity parameters for the transients are only very slightly changed by the temperature and heat flow change caused by reduced coolant flow. The primary effect is on the minimum DNBR as a direct consequence of the lowered flow (and slightly raised average temperature). The DNBR changes for the control rod withdrawal at power event can be compensated for by a change in the over temperature delta T trip setpoint. The event has been reanalyzed with this change and is satisfactory.

The peak enthalphy for the control rod ejection event is not significantly changed. The DNBR has been generically calculated using very conservative parameters and is bounding even for the low flow situation of tube plugging. These the results are satisfactory.

The control rod drop at power event has been reanalyzed and the zero power rod withdrawal event reevaluated with the reduced flow using the methodology and other assumptions of the FSAR and were found to meet the same criteria. This reanalysis is satisfactory.

III. Summary

The licensee has calculated the impact of 24% steam generator tube plugging on a postulated design basis LOCA, and on a variety of non-LOCA transients and accidents. The analyses were performed with accepted computer codes, using input parameters consistent with the FSAR. The results of the calculations are reasonable, and generally conform to our understanding of how the transients and accidents would be affected by increased tube plugging and the other proposed changes.

The licensee has presented qualitative and quantitative arguments to demonstrate that a broad spectrum of other accidents need not be recalculated as a result of the proposed changes. Those discussions presented an accurate representation of the probable impact of the proposed changes.

Although the proposed changes result in some reduction of safety margin for several of the accidents and transients, the results are clearly within the acceptance criteria of the regulations and the standard review plan. We conclude that the revised analyses form an acceptable basis for approving the proposed technical specification changes.

IV. Environmental Consideration

We have determined that the amendment does not authorize a change in effluent types or total amounts nor an increase in power level and will not result in any significant environmental impact. Having made this determination, we have further concluded that the amendment involves an action which is insignificant from the standpoint of environmental impact and, pursuant to 10 CFR §51.5(d)(4), that an environmental impact statement or negative declaration and environmental impact appraisal need not be prepared in connection with the issuance of this amendment.

V. Conclusion

We have concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (2) such activities will be conducted in compliance with the Commission's regulations and the issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public.

Date: January 13, 1984

References

1. Letter from J. P. Bayne (NYPA) to S. A. Varga (NRC), "Proposed Changes to the Technical Specifications Related to Safety Limit, Reactor Core and Control Rod and Power Distribution Limits, Shutdown Reactivity," January 13, 1983.

2. Letter from J. P. Bayne (NYPA) to S. A. Varga (NRC), "Safety Analyses for Steam Generator Tube Plugging Level of Twenty Four Percent (24% SGTP)," December 14, 1983