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Docket No. 50-286

Mr. George T. Berry, President
and Chief Operating Officer
Power Authority of the State of New York
10 Columbus Circle
New York, New York 10019

Dear Mr. Berry:

The Commission has issued the enclosed Amendment No. 40 to Facility Operating License No. DPR-64 for the Indian Point Nuclear Generating Unit No. 3. This amendment consists of changes to the Technical Specifications in response to your application transmitted by letter dated November 2, 1981.

The amendment revises the Technical Specifications to accommodate plant operation with up to 12% of the steam generator tubes plugged.

Copies of the Safety Evaluation and the Notice of Issuance are also enclosed.

Sincerely,

Original Signed By:

John O. Thoma, Project Manager
Operating Reactors Branch No. 1
Division of Licensing

Enclosures:
1. Amendment No. 40 to DPR-64
2. Safety Evaluation
3. Notice of Issuance

cc: w/enclosures
See next page

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OFFICE	ORB 1	ORB 1	ORB 1	AD:OR	OELD		
SURNAME	CParrish	JThoma/rs	SVarga	TNovak	JThoma		
DATE	11/12/81	11/12/81	11/12/81	11/12/81	11/12/81		

Mr. George T. Berry
Power Authority of the State of New York

cc: White Plains Public Library
100 Martine Avenue
White Plains, New York 10601

Mr. Charles M. Pratt
Assistant General Counsel
Power Authority of the
State of New York
10 Columbus Circle
New York, New York 10019

Ms. Ellyn Weiss
Sheldon, Harmon and Weiss
1725 I Street, N.W., Suite 506
Washington, D. C. 20006

Dr. Lawrence R. Quarles
Apartment 51
Kendal at Longwood
Kennett Square, Pennsylvania 19348

Mr. George M. Wilverding
Manager - Nuclear Licensing
Power Authority of the
State of New York
10 Columbus Circle
New York, New York 10019

Joan Holt, Project Director
New York Public Interest
Research Group, Inc.
5 Beekman Street
New York, New York 10038

Director, Technical Development
Programs
State of New York Energy Office
Agency Building 2
Empire State Plaza
Albany, New York 12223

Mr. John C. Brons, Resident Manager
Indian Point 3 Nuclear Power Plant
P. O. Box 215
Buchanan, New York 10511

Honorable George Begany
Mayor, Village of Buchanan
188 Westchester Avenue
Buchanan, New York 10511

Mr. J. P. Bayne, Senior Vice Pres.
Power Authority of the State
of New York
Columbus Circle
New York, New York 10019

Theodore A. Rebelowski
Resident Inspector
Indian Point Nuclear Generating
U. S. Nuclear Regulatory Commission
P. O. Box 38
Buchanan, New York 10511

Joyce P. Davis, Esquire
Law Department
Consolidated Edison Company of
New York Inc.
4 Irving Place
New York, New York 10003

Jeffrey C. Cohen, Esquire
New York State Energy Office
Swan Street Building
CORE 1 - Second Floor
Empire State Plaza
Albany, New York 12223

Regional Radiation Representative
EPA Region II
26 Federal Plaza
New York, New York 10007

Ezra I. Bialik
Assistant Attorney General
Environmental Protection Bureau
New York State Department of Law
2 World Trade Center
New York, New York 10047

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

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. 40
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 November 2, 1981, 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. 40, 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

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

Attachment:
Changes to the Technical
Specifications

Date of Issuance: November 13, 1981

ATTACHMENT TO LICENSE AMENDMENT NO. 40

FACILITY OPERATING LICENSE NO. DPR-64

DOCKET NO. 50-286

Revise Appendix A as follows:

Remove Pages	Insert Pages
2.1-2	2.1-2
Figure 2.1-1	Figure 2.1-1
2.3-2	2.3-2
3.10-1	3.10-1
3.10-9	3.10-9
3.10-11	3.10-11

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]

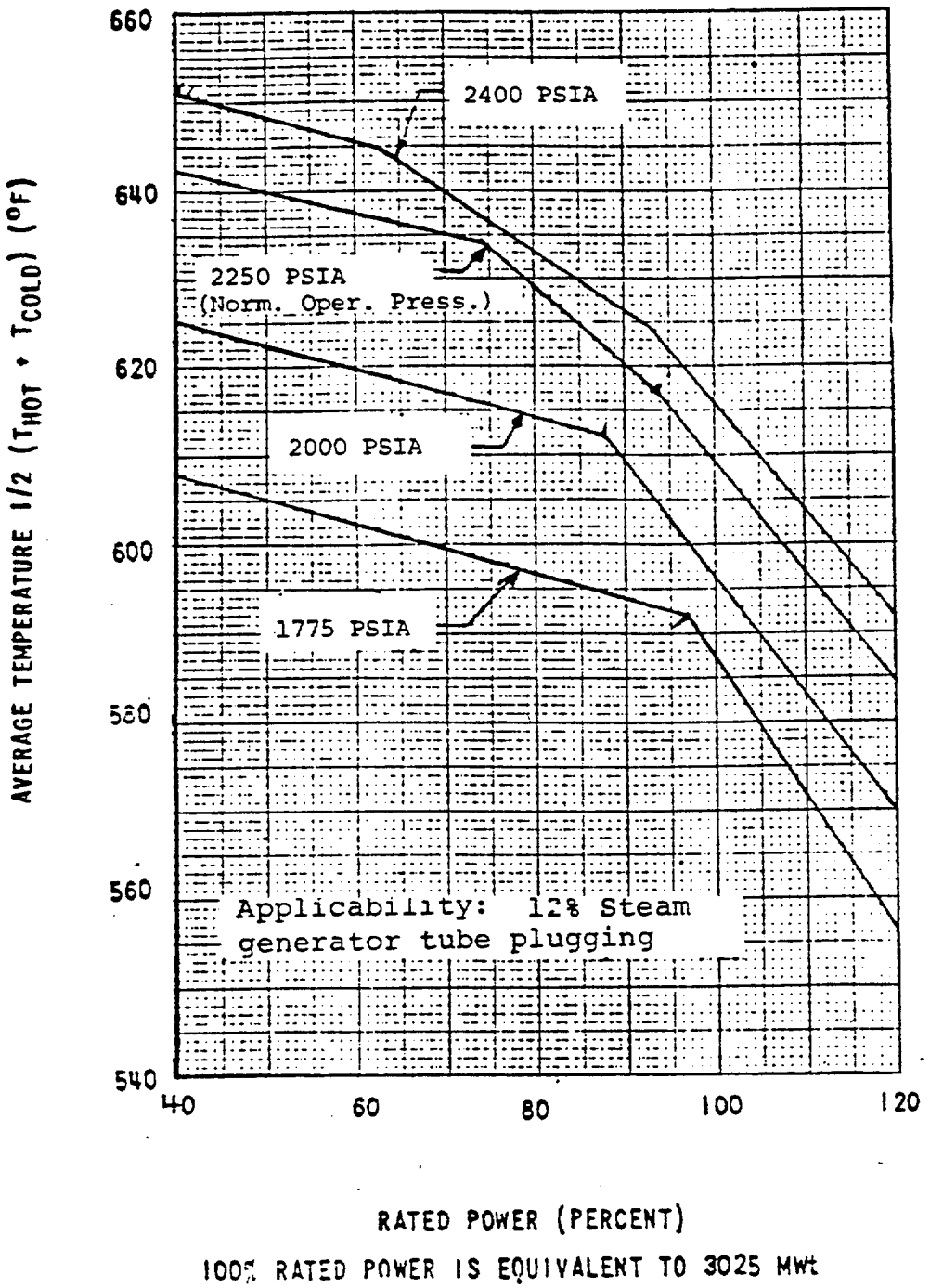

Figure 2.1-1. Core Limits - Four Loop Operation

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

$K_1 \leq 1.200$ $K_2 \geq 0.0129$ $K_3 \leq 0.00073$	Four Loop Operation	$K_1 \leq 1.110$ $K_2 \geq 0.0129$ $K_3 \leq 0.00073$	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.

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:

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

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 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 bottom 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 (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

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 40 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

BACKGROUND

The Power Authority of the State of New York (PASNY) submittal dated November 2, 1981 provides a revised emergency core cooling system (ECCS) analysis for Indian Point-3 with a steam generator tube plugging level of 12 percent. The analysis was performed using a February 1978 version of the Westinghouse evaluation model with approved modifications.

The changes to the Technical Specifications requested by the licensee to approve the 12% steam generator tube plugging limit are as follows:

a. Page 2.1-2
This page has been updated to include the new 12% steam generator tube plugging limit.

b. Figure 2.1-1
This curve has been updated to reflect the new steam generator tube plugging limit and has been corrected per Westinghouse recommendations.

c. Page 2.3-2
This page has been revised as recommended by Westinghouse to account for a slightly reduced primary coolant flow due to the increased steam generator tube plugging.

d. Page 3.10-1
This page has been updated to include the new 2.04 total core peaking factor limit as utilized in the analysis to justify a 12% steam generator tube plugging limit.

e. Page 3.10-9 and Page 3.10-11
These pages, found in the basis section of the Technical Specifications, have been updated to reflect the analysis results presented for 12% steam generator tube plugging.

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EVALUATION

The licensee (PASNY) has submitted an ECCS re-analysis assuming 12% uniform plugging of steam generator tubes. The analysis was performed with the approved version of the Westinghouse evaluation model (February 1978). Modification to computer input included reduction in the primary steam generator flow area and volume. Also included was an assessment of the impact of using the cladding swelling and rupture models proposed in NUREG-0630. Three large breaks were calculated which is appropriate for this type of re-analysis. A new value of Fq=2.04 was determined from the limiting .8 double ended break. The licensee conducted a modified final acceptance criteria analysis for their present end of cycle conditions and concluded a minimum allowed peaking factor is 2.00. Therefore the calculated value for peaking factor of 2.04 is acceptable.

The analyses were performed in accordance with 10 CFR 50 Appendix K and related RSB positions. The model changes properly reflect changes in plant conditions. The new value of Fq is properly consistent with the revised analysis. We therefore find the revised ECCS analysis and Fq changes acceptable.

The licensee has analyzed the proposed increase in steam generator tube plugging with respect to transients and accidents analyzed in the Safety Analysis Report (SAR). They have concluded that 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; and c) will not reduce the margin of safety as defined in the basis for any Technical Specification.

Based on our review of the licensee's ECCS analysis, the proposed technical specification changes are acceptable.

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.

CONCLUSION

We have concluded, based on the considerations discussed above, that: (1) because the amendment does not involve a significant increase in the probability or consequences of accidents previously considered and does not involve a significant decrease in a safety margin, the amendment does not involve a significant hazards consideration, (2) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (3) 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: November 13, 1981

UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKET NO. 50-286

POWER AUTHORITY OF THE STATE OF NEW YORK

NOTICE OF ISSUANCE OF AMENDMENT TO FACILITY OPERATING LICENSE

The U. S. Nuclear Regulatory Commission (the Commission) has issued Amendment No. 40 to Facility Operating License No. DPR-64, issued to the Power Authority of the State of New York (the licensee), which revised Technical Specifications for operation of the Indian Point Nuclear Generating Unit No. 3 (the facility) located in Buchanan, Westchester County, New York. The amendment is effective as of the date of issuance.

The amendment revises the Technical Specifications to accommodate plant operation with up to 12% of the steam generator tubes plugged.

The application for the amendment complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations. The Commission has made appropriate findings as required by the Act and the Commission's rules and regulations in 10 CFR Chapter I, which are set forth in the license amendment. Prior public notice of this amendment was not required since the amendment does not involve a significant hazards consideration.

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The Commission has determined that the issuance of this amendment will not result in any significant environmental impact and that pursuant to 10 CFR §51.5(d)(4) an environmental impact statement or negative declaration and environmental impact appraisal need not be prepared in connection with issuance of this amendment.

For further details with respect to this action, see (1) the application for amendment dated November 2, 1981, (2) Amendment No. 40 to License No. DPR-64, and (3) the Commission's related Safety Evaluation. All of these items are available for public inspection at the Commission's Public Document Room, 1717 H Street, N. W., Washington, D. C. and at the White Plains Public Library, 100 Martine Avenue, White Plains, New York. A copy of items (2) and (3) may be obtained upon request addressed to the U. S. Nuclear Regulatory Commission, Washington, D. C. 20555, Attention: Director, Division of Licensing.

Dated at Bethesda, Maryland, this 13th day of November 1981.

FOR THE NUCLEAR REGULATORY COMMISSION

Marshall Grotenhuis

Marshall Grotenhuis, Acting Branch Chief
Operating Reactors Branch No. 1
Division of Licensing