January 29, 1986

Docket No. 50-247

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Mr. John D. O'Toole
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4 Irving Place
New York, New York 10003

Dear Mr. O'Toole:

The Commission has issued the enclosed Amendment No. 109 to Facility Operating License No. DPR-26 for the Indian Point Nuclear Generating Unit No. 2. The amendment consists of changes to the Technical Specifications in response to your application transmitted by letter dated April 10, 1985.

The amendment changes the Technical Specification to revise the limits for total nuclear peaking factor (F_Q) and accumulator water volume to accommodate plant operation at steam generator plugging levels up to 25%. The change permits a maximum F_Q of 2.32 up to 25% steam generator tube plugging with nominal accumulator volumes of 822 ft^3.

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

Sincerely,

/s/MSlosson

Marylee M. Slosson, Project Manager
PWR Project Directorate #3
Division of PWR Licensing-A

Enclosures:
1. Amendment No. 109 to DPR-26
2. Safety Evaluation

cc: w/enclosures
See next page

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CVogan	MSlosson	L. Harmon	SVarga
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Indian Point Nuclear Generating
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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

CONSOLIDATED EDISION COMPANY OF NEW YORK, INC.

DOCKET NO. 50-247

INDIAN POINT NUCLEAR GENERATING UNIT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 109
License No. DPR-26

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

   A. The application for amendment by Consolidated Edison Company of New York, Inc. (the licensee) dated April 10, 1985, 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.

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-26 is hereby amended to read as follows:

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(2) Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 109 , 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 to be implemented within 30 days.

FOR THE NUCLEAR REGULATORY COMMISSION

for Steven A. Varga, Director
PWR Project Directorate #3
Division of PWR Licensing-A

Attachment:
Changes to the Technical
Specifications

Date of Issuance: January 29, 1986

ATTACHMENT TO LICENSE AMENDMENT

AMENDMENT NO.109 TO FACILITY OPERATING LICENSE NO. DPR-26

DOCKET NO. 50-247

Revise Appendix A as follows:

Remove Pages	Insert Pages
3.3-1	3.3-1
3.3-17	3.3-17
3.10-1	3.10-1
3.10-9	3.10-9
3.10-11	3.10-11
3.10-16	3.10-16
Figure 3.10-2a	Figure 3.10-2
Figure 3.10-2b	_____

3.3 ENGINEERED SAFETY FEATURES

Applicability

Applies to the operating status of the Engineered Safety Features.

Objective

To define those limiting conditions for operation that are necessary: (1) to remove decay heat from the core in emergency or normal shutdown situations, (2) to remove heat from containment in normal operating and emergency situations, (3) to remove airborne iodine from the containment atmosphere following a Design Basis Accident, (4) to minimize containment leakage to the environment subsequent to a Design Basis Accident.

Specification

The following specifications apply except during low temperature physics tests.

A. Safety Injection and Residual Heat Removal Systems

1. The reactor shall not be made critical, except for low temperature physics tests, unless the following conditions are met:

   a. The refueling water storage tank contains not less than 345,000 gallons of water with a boron concentration of at least 2000 ppm.

   b. Deleted

   c. The four accumulators are pressurized to at least 600 psig and each contains a minimum of 814.5 ft^3 and a maximum of 829.5 ft^3 of water with a boron concentration of at least 2000 ppm. None of these four accumulators may be isolated.

   d. Three safety injection pumps together with their associated piping and valves are operable.

   e. Two residual heat removal pumps and heat exchangers together with their associated piping and valves are operable.

   f. Two recirculation pumps together with the associated piping and valves are operable.

(10) Indian Point Unit No. 2, "Analysis of the Emergency Core Cooling System in Accordance with the Acceptance Criteria of 10 CFR 50.46 and 10 CFR Part 50, Appendix K", dated April, 1985.

(11) Letter from Wiliam J. Cahill, Jr. of Consolidated Edison Company of New York, to Robert W. Reid of the Nuclear Regulatory Commission, dated July 13, 1976. Indian Point Unit No. 2 Small Break LOCA Analysis.

(12) Indian Point Unit No. 3 FSAR Sections 6.2 and 6.3 and the Safety Evaluation accompanying "Application for Amendment to Operating License" sworn to by Mr. William J. Cahill, Jr. on March 28, 1977.

(13) FSAR Sections 6.5 and 6.6.

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:

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

(b) For 25% steam generator tube plugging:

$F_Q(Z) \leq (2.32/P) \times K(Z)$ for $P \geq .5$

$F_Q(Z) \leq (4.64/P) \times K(Z)$ for $P \leq .5$

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.

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

The upper bound envelope of the total peaking factor (F_Q) of specification 3.10.2.1 times the normalized peaking factor axial dependence of Figure 3.10-2 has been determined from extensive analyses considering 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 the specified F_Q times the normalized envelope of Figure 3.10-2 indicate a peak clad temperature of less than 2200°F for the worst case double-ended cold leg guillotine break.(1)

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 $F^N_{\Delta H}$ 1.55/1.08. The logic behind the larger uncertainty in this case is that (a) normal perturbations in the radial power shape (e.g., rod misalignment) affect $F^N_{\Delta H}$, in most cases without necessarily affecting F_Q, (b) the operator has a direct influence on F_Q thorugh movement of rods, and can limit it to the desired value, he has no direct control over $F^N_{\Delta H}$ and (c) an error in the predictions for radial power shape, which may be detected during startup physics tests can be compensated for in F_Q by tighter axial control, but compensation for $F^N_{\Delta H}$ is less readily available. When a measurement of $F^N_{\Delta H}$ is taken, experimental error must be allowed for and 4 percent is the appropriate allowance for a full core map taken with the moveable incore detector flux mapping system.

to limit the difference between the current value of Flux Difference (Δ I) and a reference value which corresponds to the full power equilibrium value of Axial Offset (Axial Offset = Δ 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 the total peaking factor upper bound envelope of specified F_Q 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 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 proceeds). This value, divided by the fraction of full power at which the core was operating is the full power value of the target flux difference. Values for all other core power levels are obtained by multiplying the full power value by the fractional power. Since the indicated equilibrium value was noted, no allowances for excore detector error are necessary and indicated deviation of ± 5 percent Δ I are permitted from the indicated reference value. During periods where extensive load following is required, it may be impractical to establish the required core conditions for measuring the target flux difference every month. For this reason, the specification provides two methods for updating the target flux difference. Figure 3.10-5 shows a typical construction of the target flux difference band at BOL and Figure 3.10-6 shows the typical variation of the full power value with burnup.

Strict control of the flux difference (and rod position) is not as necessary during part power operation. This is because xenon distribution control at part power is not as significant as the control at full power and allowance has been made in predicting the heat flux peaking factors for less strict control at part power. Strict control of the flux difference is not possible during certain physics tests or during required, periodic, excore calibrations which require larger flux

accident for an isolated fully inserted rod will be worse if the residence time of the rod is long enough to cause significant non-uniform fuel depletion. The 4 week period is short compared with the time interval required to achieve a significant non-uniform fuel depletion.

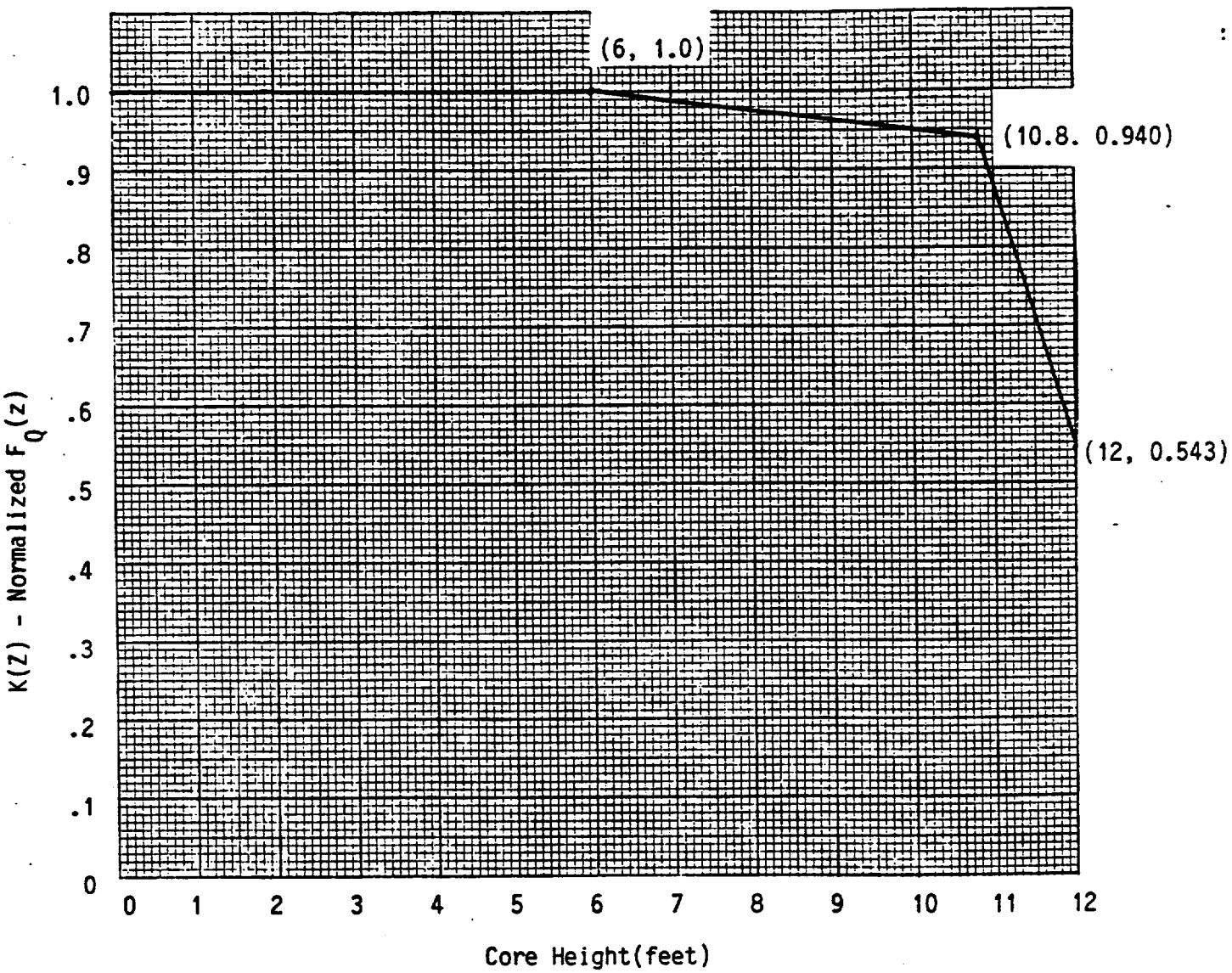
The required drop time to dashpot entry is consistent with safety analysis.

REFERENCE

1. Indian Point Unit No. 2, "Analysis of the Emergency Core Cooling System in Accordance with the Acceptance Criteria of 10 CFR 50.46 and 10 CFR Part 50, Appendix K," dated April, 1985.

Figure 3.10-2

HOT CHANNEL FACTOR NORMALIZED OPERATING ENVELOPE
(For S.G. tube plugging levels up to 25%)



Amendment No. 109

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 109 TO FACILITY OPERATING LICENSE NO. DPR-26

CONSOLIDATED EDISON COMPANY OF NEW YORK, INC.

INDIAN POINT NUCLEAR GENERATING UNIT NO. 2

DOCKET NO. 50-247

INTRODUCTION

By letter dated April 10, 1985, Consolidated Edison submitted a proposed amendment to the Technical Specifications to revise the limits for total nuclear peaking factor (F_Q) and accumulator water volume to accommodate plant operation at steam generator plugging levels up to 25%. The requested revision permits a maximum F_Q of 2.32 up to 25% steam generator tube plugging with nominal accumulator volumes of 822 Ft^3. By letter dated December 19, 1985 Consolidated Edison provided additional information supporting the amendment request.

EVALUATION

A revised large-break LOCA ECCS analysis was performed for Indian Point Unit 2. Changes in the analysis assumptions included 25% steam generator tube plugging (uniform distribution) and a nominal accumulator water volume of 822 cubic feet. The analysis also assumed a 102% power level and a total peaking factor (F_Q) of 2.32.

The large-break LOCA analysis provided by the licensee utilizes the 1981 version of the Westinghouse Evaluation model. This is the current acceptable model for licensee evaluations and satisfies the requirements of 10 CFR 50.46 and Appendix K to Part 50. The methods for accounting for steam generator tube plugging are an integral part of the Westinghouse Evaluation model and have been previously approved by the NRC.

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The large-break LOCA ECCS analyses for reactor coolant system double-ended cold leg guillotine (DECLG) breaks with discharge multipliers (C_D) of 0.6 and 0.4 were provided. Previous analyses using essentially the same input parameters but varying the accumulator water volume for optimization have demonstrated that, for 25% steam generator tube plugging, the C_D = 0.8 break size is always less limiting than the C_D = 0.6 or 0.4 cases. The limiting break size was determined to be the C_D = 0.4 DECLG break size.

The current percentage of plugged steam generator tubes range between 4.2% and 6.1%. The last inservice inspection performed by the licensee indicates that the relative variations in plugging are similar. Therefore, the uniform distribution assumption is acceptable. In addition, there is no significant asymmetry between the steam generator which would warrent additional studies to address this concern.

The licensee has determined that the proposed revision to the Technical Specification Limit for a 2.32 total peaking factor does not alter the results of previously accepted small-break LOCA analyses.

In December 1981, the NRC staff and Westinghouse held a meeting to discuss the impact of maximum safety injection on large-break LOCA analyses. For reference, a brief discussion of this issue follows:

Westinghouse ECCS analyses assume minimum safeguards for the safety injection flow, which minimizes the amount of flow to the RCS by assuming maximum injection line resistances, degraded ECCS pump performance, and the loss of one RHR pump as the most limiting single failure. This is the limiting single failure assumption when offsite power is unavailable for most Westinghouse plants. However, for some Westinghouse four loop, non-UHI, non-burst node limited plants, the current nature of the Appendix K ECCS evaluation models is such that it may be more limiting to assume the maximum possible ECCS flow delivery. In that case, maximum safeguards which assumes minimum injection line resistance, enhanced ECCS pump performance and no single failure, results in the highest amount of flow delivered to the RCS.

The discussions of this phenomenon resulted in the following agreement:

> "In future analyses, the single failure assumed will be the same as modelled currently. For four loop non-UHI non-burst node limited plants, an additional analysis will be repeated for the worst break size assuming no single failure. All cases which are analyzed will be reported to the NRC."

In accordance with this agreement the worst case large-break LOCA for Indian Point Unit 2 for this present analysis (C_D = 0.4) was re-analyzed assuming maximum safeguards. The resulting peak cladding temperature of 1904°F (versus 1944°F for the minimum safeguards) justifies the use of the minimum safeguard analysis for this evaluation.

The licensee had noted in the submittal that the effect of 25% steam generator tube plugging on non-LOCA transients was previously submitted to the NRC on April 25, 1980 and approved by the NRC via licensee amendment number 69 on April 22, 1981. In response to our concern regarding the validity of the non-LOCA design basis for the proposed 2.32 F_Q limit, the licensee converted the $F_{\Delta H}$ and F_Z factors used in the 1980 analyses to an equivalent F_Q value. The equivalent F_Q value of 2.7 bounds the proposed 2.32 F_Q limit. Since the previously accepted non-LOCA evaluation for 25% steam generator tube plugging bounds the current proposed F_Q limit, and since no appreciable asymmetric steam generator tube plugging exists between the generators, additional studies to address these concerns are not required.

FINDINGS

This revised ECCS performance evaluation has been made using an NRC approved evaluation model which satisfies the requirements of 10 CFR 50.46 and Appendix K to Part 50. A nominal accumulator water volume of 822 cubic feet, to support the 814.5 to 829.5 cubic feet Technical Specification limits, was used in the analysis. A 25% uniform steam generator tube plugging limit was assumed for the analysis. A maximum total peaking factor (F_Q) of 2.32 was used resulting in a peak cladding temperature of 1944°F for the limiting break.

The total core hydrogen generation is less than 0.3% and the local cladding oxidation is 4.17% for the limiting break. These values are within the required limits as specified in 10 CFR 50.46.

In accordance with the 1981 agreement between the NRC and Westinghouse concerning the minimum or maximum safeguards assumption, the licensee has justified the use of the minimum safeguards assumption for this evaluation.

The licensee has determined that the proposed revision to the Technical Specification Limit for a 2.32 total peaking factor does not alter the results of previously accepted small-break LOCA analyses.

The use of a uniform distribution to address steam generator tube plugging was justified based on current, actual conditions which exist at Indian Point Unit 2. The small variation between generators (4.9% to 6.1% plugging) does not warrent any additional evaluation to address the effects of asymmetric steam generator tube plugging, at this time.

The licensee has reviewed the non-LOCA design base, previously approved as license amendment number 69, and determined that these previously approved analyses bound the proposed Technical Specification changes for this current evaluation.

The proposed changes to Technical Specifications 3.3 and 3.10 are supported by the safety analyses provided by the licensee, as is required by 10 CFR 50.34.

Environmental Consideration

This amendment involves a change in the installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. The staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that this amendment involves no significant hazards consideration and there has been no public comment on such finding. Accordingly, this amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR Sec 51.22(c)(9). Pursuant to 10 CFR 51.22(b) no environmental impact statement or environmental assessment need be prepared in connection with the issuance of this amendment.

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.

Dated: JAN 2 9 1986

Principal Contributor:

E. Throm