Docket file
50-247

MARCH 20 1979

Docket No. 50-247

Mr. William J. Cahill, Jr.
Vice President
Consolidated Edison Company
of New York, Inc.
4 Irving Place
New York, New York 10003

Dear Mr. Cahill:

The Commission has issued the enclosed Amendment No. 52 to Facility Operating License No. DPR-26 for the Indian Point Nuclear Generating Unit No. 2. This amendment consists of changes to the Technical Specifications in response to your letter dated January 5, 1979. That letter provided a corrected ECCS analysis using an approved model as required by NRC Order for Modification of License dated April 27, 1978.

The amendment revises the Technical Specifications limits for the total nuclear peaking factor (F_Q), accumulator water volume and hot channel factor normalized operating envelope. Your submittal of January 5, 1979, together with this amendment, satisfies the requirements of our Order for Modification of License dated April 27, 1978. Accordingly, that Order is hereby terminated.

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

Sincerely,

Original Signed By

Cmsl
ccp

A. Schwencer, Chief
Operating Reactors Branch #1
Division of Operating Reactors

Enclosures:
1. Amendment No. 52 to DPR-26
2. Safety Evaluation
3. Notice of Issuance

7904210033

cc: w/enclosures
See next page

OFFICE	DOR:ORB1	DOR:ORB1		DOR:S&P	OELD	DOR:ORB1
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DATE	03/13/79	03/15/79		1/30/79	3/21/79	3/20/79

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Docket File

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

March 20, 1979

Docket No. 50-247

Mr. William J. Cahill, Jr.
Vice President
Consolidated Edison Company
of New York, Inc.
4 Irving Place
New York, New York 10003

Dear Mr. Cahill:

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Sincerely,

A. Schwencer

A. Schwencer, Chief
Operating Reactors Branch #1
Division of Operating Reactors

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2. Safety Evaluation
3. Notice of Issuance

cc: w/enclosures
See next page

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

CONSOLDIATED EDISON COMPANY OF NEW YORK, INC.

DOCKET NO. 50-247

INDIAN POINT NUCLEAR GENERATING UNIT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 52
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 January 5, 1979, 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-26 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 52, 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

A. Schwencer

A. Schwencer, Chief
Operating Reactors Branch #1
Division of Operating Reactors

Attachment:
Changes to the Technical
Specifications

Date of Issuance: March 20, 1979

ATTACHMENT TO LICENSE AMENDMENT NO. 52

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
3.3-15	3.3-15
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 3.10-2	Figure 3.10-2

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. The boron injection tank contains not less than 1000 gallons of a 11 1/2% to 13% by weight (20,000 ppm to 22,500 ppm of boron) boric acid solution at a temperature of at least 145°F. Two channels of heat tracing shall be available for the flow path. Valves 1821 and 1831 shall be open and valves 1822A and 1822B shall be closed, except during short periods of time when they can be cycled to demonstrate their operability.

   c. The four accumulators are pressurized to at least 600 psig and each contains a minimum of 716 ft^3* and a maximum of 731 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.

* Pending return to operation for Cycle 4 this value shall be 800 ft^3.
** Pending return to operation for Cycle 4 this value shall be 815 ft^3.

References

(1) FSAR Section 9

(2) FSAR Section 6.2

(3) FSAR Section 6.2

(4) FSAR Section 6.3

(5) FSAR Section 14.3.5

(6) FSAR Section 1.2

(7) FSAR Section 8.2

(8) FSAR Section 9.6.1

(9) FSAR Section 14.3

(10) Indian Point Unit No. 2 "Analysis of the Emergency Core Cooling System in Accordance with the Acceptance Criteria of 10CFR50.46 and Appendix K of 10CFR50", December 1978

(11) Letter from William 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.

3.10 CONTROL ROD AND POWER DISTRIBUTION LIMITS

Applicability:

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

Objectives:

To ensure:

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

Specifications:

3.10.1 Shutdown Reactivity

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

3.10.2 Power Distribution Limits

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

$$F_Q(Z) \leq (2.31^{*}/P) \times K(Z) \text{ for } P > .5$$
$$F_Q(Z) \leq (4.62^{**}) \times K(Z) \text{ for } P \leq .5$$
$$F_{\Delta H}^{N} \leq 1.55\ [1 + 0.2\ (1-P)]$$

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.

*Pending return to operation for Cycle 4 this value shall be 2.24.
**Pending return to operation for Cycle 4 this value shall be 4.48.

$F^N_{\Delta H}$, 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 2.31 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 2.31 times the normalized envelope of Figure 3.10-2 indicate a peak clad temperature of 2172.5°F for the double-ended cold leg guillotine break with C_D = 0.6, the worst case break. This corresponds to a 27.5°F margin to the 2200°F limit.[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 $F^N_{\Delta H} \leq 1.55/1.08$. 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 through 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 (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 2.31 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 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 ΔI 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

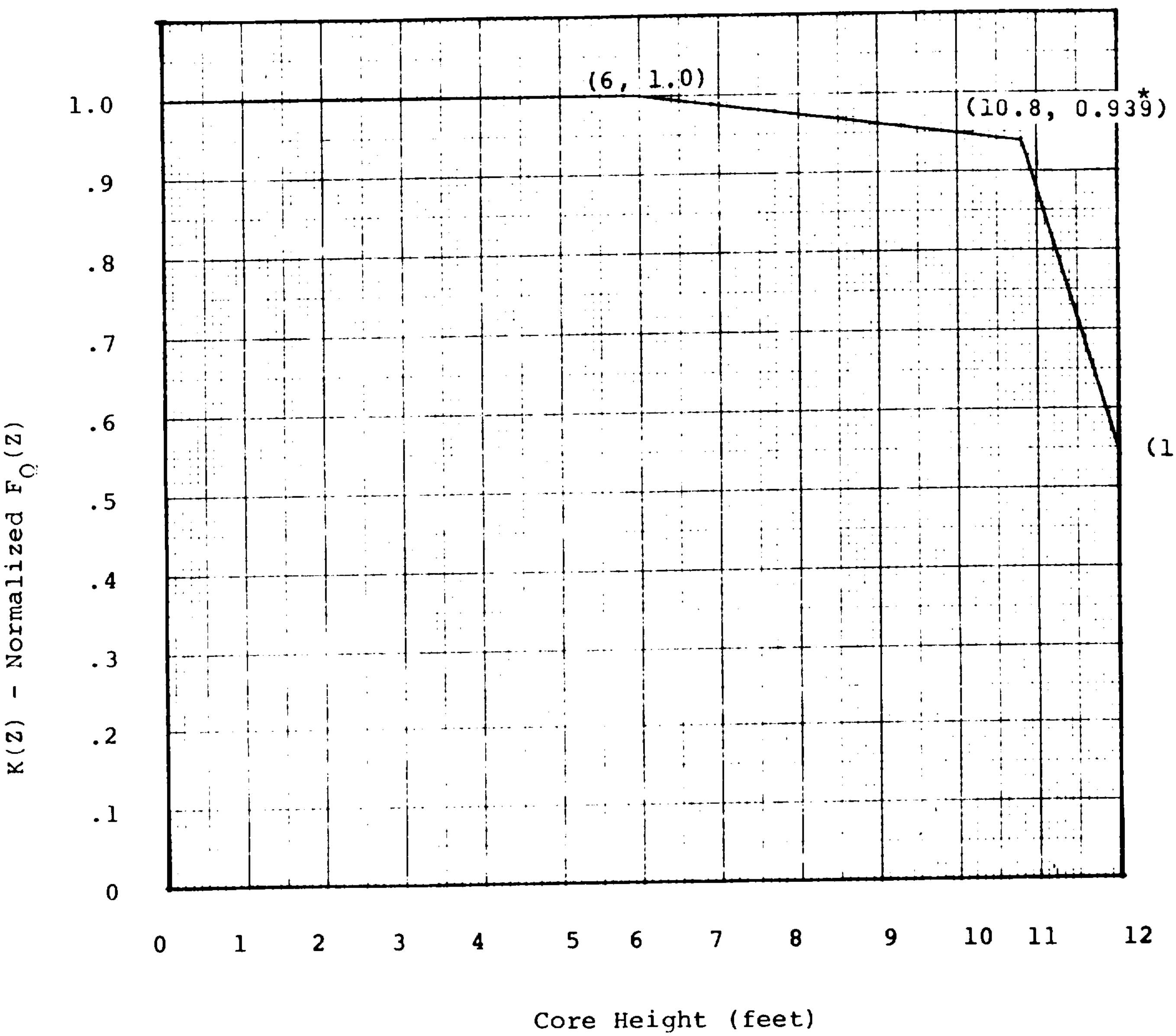
The required drop time to dashpot entry is consistent with safety analysis.

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 Appendix K of 10 CFR 50. See also Consolidated Edison Company's letter to NRC dated January 5, 1979 which submitted the results of this reanalysis based on the Westinghouse ECCS Evaluation Model approved by NRC letter to Westinghouse dated August 29, 1978.

Figure 3.10-2

HOT CHANNEL FACTOR NORMALIZED OPERATING ENVELOPE



*Pending return to operation for Cycle 4 this value shall be 0.94.
**Pending return to operation for Cycle 4 this value shall be .54.

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 52 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

In response to an NRC Order for Modification of License issued April 27, 1978 (see below), on January 5, 1979 (Reference 1), Consolidated Edison Company (the licensee) proposed changes to the Technical Specifications of Facility Operating License DPR-26 for Indian Point Nuclear Generating Unit 2. The proposed changes consist of revisions of the required accumulator tank volume and the total nuclear peaking factor (F_Q), and modification of hot channel factor normalized operating envelope (Figure 3.10-2 in the Technical Specifications). The licensee justified the proposed changes by providing a reanalysis of ECCS using a recently modified Westinghouse evaluation model (References 2 and 3). This model has been reviewed and approved by the staff (Reference 4). It includes the correction for the Zr-Water reaction error.

Presently, Indian Point Unit 2 is operating with the interim value of total peaking factor of 2.24 (Reference 5). This value was imposed by the Order for Modification of License (Reference 6) after an error in the heat generated by Zr-water reactor had been discovered in the Westinghouse ECCS evaluation model. The order requested the licensee to submit, as soon as possible, a reevaluation of the ECCS performance calculated in accordance with the corrected and approved evaluation model. The present submittal fulfills this requirements.

Evaluation

The licensee analyzed the ECCS performance for a large break LOCA using the modified Westinghouse evaluation model. The analysis was performed for a double ended guillotine cold leg break (DECLG) with discharge coefficients of C_D=1.0, 0.8, 0.6 and 0.4. The analysis conservatively assumes 6 percent uniform steam generator tube plugging. Actual plugging is less than 4 percent on all steam generators.

Input parameters assumed in the analysis are listed below:

Core Power: 102 percent of 2758 MWt (rated power)
Peak Linear Power: 103 percent of 13.36 kW/ft
Peaking Factor: 2.31
Accumulator Water Volume: 716 ft^3 per accumulator

The analysis shows that the C_D=0.6 break results in the highest peak clad temperature. The results of the ECCS analysis indicate a peak cladding temperature of 2173°F, a maximum local Zr-water reactor of 6.14 percent and a total Zr-water temperature reaction of less than 0.3 percent. All these values are within the allowable limits specified in 10 CFR 50.46.

The licensee did not include small break analysis since neither steam generator tube plugging nor correction of the Zr-water error significantly affect the results of this analysis.

A subset of the "18 case" analysis was provided by the licensee (Reference 5) to justify operation with an F_Q limit of 2.24 without additional incore monitoring. This is conservative with respect to the F_Q value used in the updated ECCS analysis for Indian Point Unit 2.

We concur, therefore, with the following proposed changes to the Technical Specifications:

(1) Total peaking factor, F_Q, shall not exceed a maximum limit of F_Q=2.31,

(2) Accumulator water volume shall be reduced to 716 cu ft, minimum and 731 cu ft maximum, and

(3) The hot channel factor normalized operating envelope shall be modified.

However, to change the accumulator water volume requires entering containment. The licensee has proposed to make this change during the next refueling outage, presently scheduled to begin in June 1979. We find that continued operation for this interim period with the current accumulator water volumes is acceptable provided the F_Q remains limited to 2.24. Asterisks have been provided in the Technical Specifications to indicate the limits for the accumulator water volume, F_Q, and the hot channel factor normal operating envelope that shall be used for this interim period.

Summary

Based on the review of the submitted documents, we conclude that the results of the ECCS reanalysis, performed with the February 1978 version of the Westinghouse ECCS evaluation model corrected for Zr-water reaction error and including the assumption of 6 percent uniform steam generator tube plugging, yield values of LOCA parameters which are conservative relative to the 10 CFR 50.46 criteria. We consider the submitted ECCS reanalysis and the resultant changes to the Technical Specifications acceptable for operation of the plant with up to 6 percent steam generator tubes plugged.

Environmental Consideration

We have determined that this amendment does not authorize a change in effluent types or total amounts or an increase in power level and will not result in any significant environmental impact. Having made this determination, we have further concluded that this 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 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 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.

Dated: March 20, 1979

References

1. Consolidated Edison Company of New York (CECNY) letter (Cahill) to NRC dated January 5, 1979.

2. WCAP-9220, Westinghouse ECCS Evaluation Model, February 1978 Version, WCAP-9220-P-A (Proprietary), WCAP-9221-A (Non-Proprietary), February, 1978.

3. Westinghouse letter (Eicheldinger) to NRC (Stolz) dated April 7, 1978.

4. NRC letter (Stolz) to Westinghouse (Anderson) dated August 29, 1978.

5. CECNY letter (Cahill) to NRC (Schwencer) dated April 17, 1978.

6. NRC letter (Schwencer) to CECNY (Cahill) dated April 27, 1978.

UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKET NO. 50-247

CONSOLIDATED EDISON COMPANY OF NEW YORK, INC.

NOTICE OF ISSUANCE OF AMENDMENT TO FACILITY OPERATING LICENSE

AND TERMINATION OF AN OUTSTANDING ORDER

The U. S. Nuclear Regulatory Commission (the Commission) has issued Amendment No. 52 to Facility Operating License No. DPR-26 issued to Consolidated Edison Company of New York, Inc. (the licensee), which revised Technical Specifications for operation of the Indian Point Nuclear Generating Unit No. 2 (the facility) located in Buchanan, Westchester County, New York. The amendment is effective as of the date of issuance.

The amendment revises the Technical Specification limits for total nuclear peaking factor (F_Q), accumulator water volume and hot channel factor normalized operating envelope.

The Commission also terminated its Order for Modification of License dated April 27, 1978 having determined that, upon issuance of this amendment, the requirements of that Order had been satisfied.

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.

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 January 5, 1979, (2) Amendment No. 52 to License No. DPR-26, (3) the Commission's related Safety Evaluation and (4) the Commission's Order for Modification of License dated April 27, 1978. 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), (3) and (4) may be obtained upon request addressed to the U. S. Nuclear Regulatory Commission, Washington, D. C. 20555, Attention: Director, Division of Operating Reactors.

Dated at Bethesda, Maryland, this 20thday of March, 1979.

FOR THE NUCLEAR REGULATORY COMMISSION

A. Schwencer

A. Schwencer, Chief
Operating Reactors Branch #1
Division of Operating Reactors