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

March 24, 1992

Docket No. 50-286

Mr. Ralph E. Beedle
Executive Vice President - Nuclear Generation
Power Authority of the State of New York
123 Main Street
White Plains, New York 10601

Dear Mr. Beedle:

SUBJECT: ISSUANCE OF AMENDMENT FOR INDIAN POINT NUCLEAR GENERATING UNIT NO. 3 (TAC NO. M81585)

The Commission has issued the enclosed Amendment No. 112 to Facility Operating License No. DPR-64 for the Indian Point Nuclear Generating Unit No. 3. The amendment consists of changes to the Technical Specifications in response to your application transmitted by letter dated August 30, 1991.

The amendment revises Technical Specifications Section 3.10 (Control Rod and Power Distribution Limits), and the Bases to allow the shutdown margin to be a constant value of 1.3 percent delta k/k. In addition, the amendment also revises Technical Specifications Section 1.0 (Definitions) to add a definition for shutdown margin.

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

Sincerely,

N. F. Conicella

Nicola F. Conicella, Project Manager
Project Directorate I-1
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 112 to DPR-64
2. Safety Evaluation

cc w/enclosures:
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EXEMPTION AUTHORIZATIONS/ORDERS

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Mr. Ralph E. Beedle
Power Authority of the State
of New York

Indian Point Nuclear Generating
Station Unit No. 3

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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. 112
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 August 30, 1991, 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-64 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. 112 , 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 and shall be implemented within 30 days.

FOR THE NUCLEAR REGULATORY COMMISSION

Robert A. Capra

Robert A. Capra, Director
Project Directorate I-1
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical Specifications

Date of Issuance: March 24, 1992

ATTACHMENT TO LICENSE AMENDMENT NO. 112

FACILITY OPERATING LICENSE NO. DPR-64

DOCKET NO. 50-286

Revise Appendix A as follows:

Remove Pages	Insert Pages
"List of Figures"	vi
1-6	1-6
3.10-1	3.10-1
3.10-2	3.10-2
3.10-5	3.10-5
3.10-15	3.10-15
3.10-16	3.10-16
Figure 3.10-1	"Figure 3.10-1 Deleted"

FIGURE 3.10-1

HAS BEEN

DELETED

LIST OF FIGURES

Title	Figure No.
Core Limits - Four Loop Operation	2.1-1
Maximum Permissible T_{cold} for First RCP Start (OPS Operable, Hottest SG Temp. > T_{cold})	3.1.A-1
Maximum Permissible RCS Pressure for RCP Start(s) with OPS Inoperable (SG Temp. > T_{cold} for additional pump starts, SG Temp. < T_{cold} for all pump starts)	3.1.A-2
RCS Pressure Limits for Low Temperature Operation	3.1.A-3
Maximum Pressurizer Level for OPS Inoperable and First RCP Start (SG Temp. > T_{cold})	3.1.A-4
Maximum Pressurizer Level with OPS Inoperable and One (1) Charging Pump Energized	3.1.A-5
Maximum Pressurizer Level with OPS Inoperable and One (1) Safety Injection Pump and/or Three (3) Charging Pumps Energized	3.1.A-6
Reactor Coolant System Heatup Limitations	3.1-1
Reactor Coolant System Cooldown Limitations	3.1-2
Primary Coolant Specific Activity Limit vs. Percent of Rated Thermal Power	3.1-3
Gross Electrical Output - 1" HG Backpressure	3.4-1
Gross Electrical Output - 1.5" HG Backpressure	3.4-2
Limiting Fuel Burnup vs. Initial Enrichment	3.8-1
Minimum Burnup for Storage of Fuel in Max Density Spent Fuel Pit Racks	3.8-2
Maximum Density Spent Fuel Pit (SFP) Racks, Regions and Indexing	3.8-3
DELETED	3.10-1
Steam Generator Primary Side Ultrasonic Test Sectors	4.2-1
Pressure/Temperature Limitations for Hydrostatic Leak Test	4.3-1

1.16 REPORTABLE EVENT

A REPORTABLE EVENT shall be any of those conditions specified in Section 50.73 to 10 CFR 50.

1.17 CORE OPERATING LIMITS REPORT

The CORE OPERATING LIMITS REPORT (COLR) is the unit-specific document that provides core operating limits for the current operating reload cycle. These cycle-specific core operating limits shall be determined for each reload cycle in accordance with Specification 6.9.1.6. Plant operation within these operating limits is addressed in individual specifications.

1.18 SHUTDOWN MARGIN

SHUTDOWN MARGIN (SDM) is the instantaneous amount of negative reactivity by which the reactor is subcritical or would be subcritical from its present condition assuming all full-length rod cluster assemblies (shutdown and control) are fully inserted except for the single rod cluster assembly of highest reactivity worth which is assumed to be fully withdrawn.

3.10 CONTROL ROD AND POWER DISTRIBUTION LIMITS

Applicability:

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

Objectives:

To ensure:

1. Core subcriticality after reactor trip.

2. Acceptable core power distribution during power operation in order to 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

3.10.1.1 Whenever $T_{avg} > 200°F$ the shutdown margin shall be $\geq 1.3\%\ \Delta k/k$.

3.10.1.2 When the conditions of specification 3.10.1.1 are not met, initiate boration to restore shutdown margin within limit.

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:

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

$$F_Q(Z) \leq (F_Q^{RTP}/0.5) \times K(Z) \text{ for } P \leq 0.5$$

$$F_{\Delta H}^N \leq F_{\Delta H}^{RTP}(1 + PF_{\Delta H}(1-P))$$

Where P is the fraction of full power at which the core is operating, K(Z) is the fraction specified in the

Amendment No. ~~13~~, ~~40~~, ~~48~~, ~~61~~, ~~73~~, ~~88~~, ~~103~~, 112

COLR, Z is the core height location of F_Q, F_Q^{RTP} is the F_Q limit at Rated Thermal Power (RTP) specified in the COLR, $F_{\Delta H}^{RTP}$ is the $F_{\Delta H}$ limit at Rated Thermal Power specified in the COLR, and $PF_{\Delta H}$ is the Power Factor Multiplier specified in the COLR.

3.10.2.2 Following initial core loading, subsequent reloading and at regular effective full power monthly intervals thereafter, power distribution maps, using the movable detector system, shall be made to confirm that the hot channel factor limits of this specification are satisfied. For the purpose of this comparison,

3.10.2.2.1 The measurement of total peaking factor F_Q^{Meas}, shall be increased by three percent to account for manufacturing tolerances and further increased by five percent to account for measurement error.

3.10.2.2.2 When $F_{\Delta H}^N$ is measured, no additional allowances are necessary prior to comparison with the limits of section 3.10.2. An error allowance of 4% has been included in the limits of section 3.10.2. If either measured hot channel factor exceeds its limit specified under Item 3.10.2.1, the reactor power and high neutron flux trip setpoint shall be reduced so as not to exceed a fraction of rated power equal to the ratio of the F_Q or $F_{\Delta H}^N$ limit to measured value, whichever is less. If subsequent incore mapping cannot, within a 24-hour period, demonstrate that the hot channel factors are met, the reactor shall be brought to a hot shutdown condition with return to power authorized only for the purpose of physics testing.

3.10.2.3 The reference equilibrium indicated axial flux difference for each excore channel as a function of power level (called the target flux difference) shall be measured at least once per equivalent full power quarter. The target flux differences must be updated each effective full power month by linear interpolation using the most recent measured value and a value of 0 percent at the end of the cycle life.

3.10.2.4 Except during physics tests, during excore calibration procedures and except as modified by Items 3.10.2.5 through 3.10.2.7 below, the indicated axial flux difference of all but one operable excore channel shall be maintained within the band specified in the COLR about the target flux difference.

3.10.3.3 The rod position indicators shall be monitored and logged once each shift to verify rod position within each bank assignment.

3.10.3.4 The tilt deviation alarm shall be set to annunciate whenever the excore tilt ratio exceeds 1.02. If one or both of the quadrant power tilt monitors is inoperable, individual upper and lower excore detector calibrated outputs shall be logged once per shift and after a load change greater than 10 percent of rated power.

3.10.4 <u>Rod Insertion Limits</u>

3.10.4.1 The shutdown rods shall be fully withdrawn as specified in the COLR when the reactor is critical or approaching criticality (i.e., the reactor is no longer subcritical by an amount equal to or greater than the shutdown margin of Specification 3.10.1).

3.10.4.2 When the reactor is critical, the control banks shall be limited in physical insertion to the insertion limits specified in the COLR.

3.10.4.3 Control bank insertion shall be further restricted if:

a) The measured control rod worth of all rods, less the worth of the most reactive rod (worst case stuck rod), is less than the reactivity required to provide the design value of available shutdown.
b) A rod is inoperable (Specification 3.10.7).

3.10.4.4 Control rod insertion limits do not apply during physics tests or during periodic exercise of individual rods. However, the shutdown margin required by Specification 3.10.1 must be maintained except for the low power physics test to measure control rod worth and shutdown margin. For this test, the reactor may be critical with all but one control rod inserted.

If tilt ratio greater than 1.09 occurs which is not due to a misaligned rod, the reactor shall be brought to a hot shutdown condition for investigation. However, if the tilt condition can be identified as due to rod misalignment, operation can continue at a reduced power (3% for each one percent the tilt ratio exceeds 1.0) for two hours to correct the rod misalignment.

A sufficient shutdown margin insures that: 1) the reactor can be made subcritical from all operating conditions, 2) the reactivity transients associated with postulated accident conditions are controllable within acceptable limits, and 3) the reactor will be maintained sufficiently subcritical to preclude inadvertent criticality in the shutdown condition.

Shutdown margin requirements vary throughout core life as a function of fuel depletion, RCS boron concentration, and RCS T_{avg}. The most restrictive condition occurs at end of life (EOL), with T_{avg} at no load operating temperature, and is associated with a postulated steam line break accident resulting in uncontrolled RCS cooldown. In the analysis of this accident, a minimum shutdown margin of 1.3 % $\Delta k/k$ is required to control the reactivity transient. Accordingly, the shutdown margin requirement is based upon this limiting condition and is consistent with FSAR safety analysis assumptions.

The action to be taken when shutdown margin in out of limit is to borate using the best available source. In the determination of the required combination of boration flow rate and boron concentration, there is no unique Design Basis Event which must be satisfied. It is imperative to raise the boron concentration of the Reactor Coolant System as soon as possible. Therefore, the operator should begin boration with the best possible source available for the plant condition.

Rod insertion limits are used to assure adequate trip reactivity, to assure meeting power distribution limits, and to limit the consequence of a hypothetical rod ejection accident. The available control rod reactivity, or excess beyond needs, decreases with decreasing boron concentration because the negative reactivity required to reduce the core power level from full power to zero is largest when the boron concentration is low.

The intent of the test to measure control rod worth and shutdown margin (Specification 3.10.4) is to measure the worth of all rods less the worth of the worst case for an assumed stuck rod, that is, the most reactive rod. The measurement would be anticipated as part of the initial startup program and infrequency over the life of the plant, to be associated primarily with determinations of special interest such as end of life cooldown, or startup of fuel cycles which deviate from normal equilibrium conditions in terms of fuel loading patterns and anticipated control bank worth. These measurements will augment the normal fuel cycle design calculations and place the knowledge of shutdown capability on a firm experimental as well as analytical basis.

The rod position indicator channel is sufficiently accurate to detect a rod ±7 inches away from its demand position. An indicated misalignment less than 12 steps does not exceed the power peaking factor limits. If the rod position indicator channel is not operable, the operator will be fully aware of the inoperability of the channel, and special surveillance of core power tilt indications, using established procedures and relying on excore nuclear detectors, and/or moveable incore detectors, will be used to verify power distribution symmetry. These indirect measurements do not have the same resolution if the bank is near either end of the core, because a 12 step misalignment would have no effect on power distribution. Therefore, it is necessary to apply the indirect checks following significant rod motion.

One inoperable control rod is acceptable provided that the power distribution limits are met, trip shutdown capability is available, and provided the potential hypothetical ejection of the inoperable rod is not worse than the cases analyzed in the safety analysis report. The rod ejection 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 5 day 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. WCAP-8576, "Augmented Startup and Cycle 1 Physics Program:, August 1975
2. FSAR Appendix 14C
3. Letter from J.P. Bayne to S.A. Varga dated April 23, 1985, entitled "Proposed Technical Specifications Regarding the Cycle 4/5 Refueling".

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 112 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

1.0 INTRODUCTION

By letter dated August 30, 1991, the Power Authority of the State of New York (the licensee) submitted a request for changes to the Indian Point Nuclear Generating Unit No. 3 (IP3), Technical Specifications (TS). The requested changes would revise Section 3.10 (Control Rod and Power Distribution Limits), and the Bases to allow the shutdown margin to be a constant value of 1.3 percent delta k/k. In addition, the amendment would also revise Section 1.0 (Definitions) to add a definition for shutdown margin.

The constant minimum shutdown margin of 1.3 percent delta k/k would be used throughout each cycle, instead of a shutdown margin which varied from 1.0 percent delta k/k at the beginning of cycle (BOC) to 1.72 percent delta k/k at the end of cycle (EOC). The licensee requested this change in the shutdown margin (SDM) to support future reload designs having increased enrichments and higher burnups, and thus longer cycle length.

The licensee made conclusions from the results of individual reviews performed by the Westinghouse Electric Corporation in such areas as loss of coolant accidents (LOCA), non-LOCA transients, containment integrity, postulated steamline break accidents and other pertinent systems effected by any change in the shutdown margin at EOC. In performing these analyses, Westinghouse used the latest approved 3-dimensional isotopic burn codes incorporating minimum uncertainties in the cross-section and other physics parameters. The 1.72 percent delta k/k value that was based on 2-dimensional isotopic burn codes was inflated because of large uncertainties. Westinghouse also considered the current licensing basis analyses, which include the most bounding effects of up to 24 percent uniform and asymmetric plugging of the steam generator tubes.

2.0 EVALUATION

2.1 Non-LOCA Evaluations

The shutdown margin analyses as reported in Chapter 14 of the final safety analysis report (FSAR) are based on two explicit assumptions: (a) the zero

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power cases of the excessive heat removal due to feedwater system malfunctions in Section 14.1.10, and (b) the analyses for rupture of steam pipe in Section 14.2.5. The following is a review of these two analyses by the licensee.

2.1.1 Zero Power Cases of Excessive Heat Removal Caused by Feedwater System Malfunctions

The FSAR provides analysis of this event for two zero power cases with the reactor critical: (1) automatic rod control, and (2) manual rod control. In each case, zero power is assumed in calculating the maximum reactivity insertion rate that would result from a malfunction in the feedwater system. The malfunction assumed is the accidental opening of one of the feedwater control valves, which would lead to an increase in feedwater flow to one of the steam generators causing a decrease in reactor coolant temperature. The reactor will respond to this temperature decrease by inserting positive reactivity, resulting from the large negative moderator temperature coefficient assumed in the analysis.

The analyses for IP3 showed that the maximum reactivity insertion rate occurred before the reactor attained peak power and did not cause the reactor to trip. Since a reactor trip does not occur, the magnitude of the shutdown margin becomes irrelevant. Consequently, the proposed change of the shutdown margin in the TS will not affect the results of the zero power cases.

2.1.2 Rupture of Steam Pipe

The licensee considered three cases in investigating the rupture of a steam pipe based on the new constant shutdown margin and its effects on the departure from nucleate boiling (DNB). Two of the cases were hypothetical (one case with offsite power and the other without). The third was a credible steam pipe break, which included the assumption of a steam release equivalent to the steam released from a steam generator safety valve. The steamline break would lead to a secondary mass release from one steam generator causing a decrease in reactor coolant temperature. Once again, reactor power will increase due to the negative temperature coefficient. In performing each of these three analyses, the licensee used a shutdown margin of 1.3 percent delta k/k. The licensee also evaluated the response of the containment pressure in other hypothetical cases using conservative assumptions to maximize the rates at which mass and energy would be released. In all cases, the licensee assumed hot shutdown initial conditions at EOC (the minimum required EOC shutdown margin and a negative moderator temperature coefficient).

The analyses showed that the hypothetical cases and the credible case met all the applicable DNB criteria for the predicted increases in reactivity and the resulting conditions. The licensee also demonstrated that the peak containment pressure is below the limit value of 47 psig.

2.1.3 Chemical and Volume Control System Malfunction

The licensee conducted analysis to determine the minimum amount of time that the operator would need to prevent a total loss of shutdown margin caused by an inadvertent dilution of the boron concentration in the reactor coolant system (RCS). The boron could be diluted because of a malfunction of the chemical and volume control system or an operator error.

The licensee analyzed this event for only those operating conditions in which the average reactor coolant temperature (T_{avg}) is over 200 °F, since the changes to the technical specifications minimum shutdown margin only apply to those operating conditions. Those operating conditions are power, reactor critical, and hot-shutdown. The results of the analyses for the power and reactor critical operating conditions were based on determining the initial boron concentration of the RCS. In each case, the licensee assumed a boron concentration that would be associated with a critical, hot zero power condition with the control rods fully inserted. The licensee found that the calculated response time for an inadvertent boron dilution event is independent of the shutdown margin for these operating conditions, since the reactor is assumed to be critical when the event begins. Consequently, a change in the minimum shutdown margin does not affect the calculated operator action time for these operating conditions.

The licensee addressed the inadvertent boron dilution event for the hot shutdown operating condition through application of the Westinghouse boron dilution interim operating procedure (Ref. 1). Application of this procedure ensures that a time period of at least 15 minutes is available from the beginning of a dilution event until a loss of shutdown margin is experienced. The licensee uses the Westinghouse guidance when developing plant specific boron concentration graphs. These graphs are referenced by the appropriate plant operating procedures. Since these guidelines are independent of shutdown margin, they remain unaffected by a change in the minimum required shutdown margin.

2.2 LOCA Evaluation

For small break LOCA, residual fissions based on a four percent shutdown margin are used as part of the decay heat model. The nature of the modeling is such that use of a four percent shutdown margin along with other features of the model provide an overall conservatism in terms of calculated peak clad temperature (PCT). This four percent shutdown margin is based on core voiding resulting from the LOCA. The change in the operational shutdown margin will not adversely affect the inherent conservatism of the model, therefore, small break LOCA results are not adversely affected. Shutdown margin is not explicitly modeled for the analyses of large break LOCA, post-LOCA long-term core cooling, hot leg switchover, and LOCA hydraulic forces.

2.3 Containment Integrity

The licensee had previously performed a containment margin improvement analysis to determine the effect of increasing service water temperature on the containment integrity. A shutdown margin at 1.3 percent delta k/k was assumed in the analysis. This analysis was documented in WCAP-12269, "Containment Margin Improvement Analysis for Indian Point 3," May 1989 (Westinghouse Proprietary). In the NRC safety evaluation on this topic (Ref. 2), the NRC concluded that the revised containment analysis yielded an acceptable containment peak pressure and the containment maximum temperature was less than the equipment qualification temperature envelope.

2.4 Assessment of No Unreviewed Safety Question

The licensee reviewed and analyzed the safety significance of the proposed change in the shutdown margin using a constant value of 1.3 percent delta k/k. The licensee performed this evaluation using the criteria specified in Section 50.59 of Title 10 of the Code of Federal Regulations (10 CFR 50.59). This analysis does not present an unreviewed safety question.

3.0 SUMMARY

The staff has reviewed and analyzed the data, methodology, and conclusion of the licensee and finds they are appropriate and acceptable. The licensee has accounted for all licensing basis accidents which consider a shutdown margin of 1.3 percent delta k/k and has demonstrated acceptable results. The changes will not constitute an unreviewed safety question and will not adversely affect the safety of plant operations. Therefore, the shutdown margin change involves no significant hazards consideration. Consequently, the proposed changes are acceptable.

4.0 STATE CONSULTATION

In accordance with the Commission's regulations, the New York State official was notified of the proposed issuance of the amendment. The State official had no comments.

5.0 ENVIRONMENTAL CONSIDERATION

The amendment changes a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. The NRC 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 the amendment involves no significant hazards consideration, and there has been no public comment on such finding (56 FR 49925). Accordingly, the amendment

meets the eligibility criteria for categorical exclusion set forth in 10 CFR 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 the amendment.

6.0 CONCLUSION

The Commission has 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, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

7.0 REFERENCES

1. Westinghouse letter 89IN*-G-0018, "New York Power Authority Indian Point Unit 3, Boron-Dilution Interim Operating Procedure," February 20, 1989.

2. "Safety Evaluation by the Office of Nuclear Reactor Regulation Related to Amendment No. 98 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," NRC, May 7, 1990.

Principal Contributor:
A. Attard

Date: March 24, 1992

March 24, 1992

Docket No. 50-286

DISTRIBUTION:
See attached sheet

Mr. Ralph E. Beedle
Executive Vice President - Nuclear Generation
Power Authority of the State of New York
123 Main Street
White Plains, New York 10601

Dear Mr. Beedle:

SUBJECT: ISSUANCE OF AMENDMENT FOR INDIAN POINT NUCLEAR GENERATING UNIT NO. 3 (TAC NO. M81585)

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

Sincerely,
ORIGINAL SIGNED BY:
Nicola F. Conicella, Project Manager
Project Directorate I-1
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 112 to DPR-64
2. Safety Evaluation
cc w/enclosures:
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DATE	: 3/3/92	:3/3/92	:3/5/92	:3/24/92	:

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DATED: March 24, 1992

AMENDMENT NO. 112 TO FACILITY OPERATING LICENSE NO. DPR-64-INDIAN POINT UNIT 3

Docket File
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PDI-1 Reading
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OGC-WF
D. Hagan, 3302 MNBB
G. Hill (4), P-137
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C. Grimes, 11/F/23
R. Jones, 8E21
A. Attard 8/E/23
ACRS (10)
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OC/LFMB
PD Plant-specific file

cc: Plant Service list